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Shutdown Decay Heat Removal Analysis of a Combustion Engineering 2-Loop Pressurized Water Reactor

Case Study

COMPLETED

Prepared by W. R. Cramond, D. M. Ericson, Jr., G. A. Sanders

Sandia National Laboratories

Prepared for
U.S. Nuclear Regulatory
Commission

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ABSTRACT

This is one of six case studies for USI A-45 Decay Heat Removal (DHR) Requirements. The purpose of this study is to identify any potential vulnerabilities in the DHR systems of a typical Combustion Engineering 2-loop PWR, to suggest possible modifications to improve the DHR capability, and to assess the value and impact of the most promising alternatives to the existing DHR systems. The systems analysis considered small LOCAs and transient internal initiating events, and seismic, fire, extreme wind, internal and external flood, and lightning external events. A full-scale systems analysis was performed with detailed fault trees and event trees including support system dependencies. The system analysis results were extrapolated into release categories using applicable past PRA phenomenological results and improved containment failure mode probabilities. Public consequences were estimated using site specific CRAC2 calculations. The Value-Impact (VI) analysis of possible alternatives considered both onsite and offsite impacts arriving at several risk measures such as averted population dose out to a 50-mile radius and dollars per person rem averted. Uncertainties in the VI analysis are discussed and the issues of feed and bleed and secondary blowdown are analyzed.

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iv

TABLE OF CONTENTS

Page

List of Figures ix
List of Tables x

1.0 INTRODUCTION 1-1 1
1.1 Objectives 1-1 1
1.2 Background 1-1 1
1.3 The St. Lucie Nuclear Plant 1-2 2
1.4 The Shutdown Decay Heat Removal Analysis . . . 1-8 8

19
2.0 INTERNAL ANALYSIS - METHODS AND RESULTS 2-1 19
2.1 Analysis Methods 2-1 19
2.1.1 Modeling 2-1 19
2.1.2 Accident Sequence Analysis 2-4 22
2.1.3 Vulnerability Identification 2-5 23
2.1.4 Containment System Integration 2-5 23
2.1.5 Interface with Accident Phenomenology. . . 2-6 24
2.1.6 Value Analysis 2-6 24
2.1.7 Example 2-7 25
2.2 Core Melt Probability Results 2-8 26
2.3 Potential Internal Vulnerabilities and Modifications 2-10 28
2.4 Summary of Internal Analysis Recommended . . . Modifications 2-18 36

3.0 SPECIAL EMERGENCY ANALYSES - METHODS AND RESULTS . 3-1 37
3.1 Seismic Analysis 3-7 43
3.1.1 Analysis Methods 3-7 43
3.1.2 Seismic Vulnerability 3-11 47
3.1.3 Potential Modifications 3-13 49
3.2 Fire Analysis 3-13 49
3.2.1 Analysis Methods 3-15 51
3.2.2 Fire Vulnerability 3-17 53
3.2.3 Potential Modifications 3-19 55
3.3 Internal Flood Analysis 3-21 57
3.3.1 Analysis Methods 3-22 58
3.3.2 Internal Flood Vulnerability 3-24 60
3.3.3 Potential Modifications 3-28 64
3.4 External Flood Analysis 3-28 64
3.4.1 Analysis Methods 3-29 65
3.4.2 External Flood Vulnerability 3-31 67
3.4.3 Potential Modifications 3-40 76
3.5 Extreme Wind Analysis 3-41 77
3.5.1 Analysis Methods 3-41 77
3.5.2 Extreme Wind Vulnerability 3-42 78
3.5.3 Potential Modifications 3-46 82
3.6 Lightning Analysis 3-47 83
3.6.1 Analysis Methods 3-47 83
3.6.2 Lightning Vulnerability 3-49 85
3.6.3 Potential Modifications 3-52 88

3.7 Sabotage Analysis 3-52 88
3.8 Summary of Special Emergency Modifications . . 3-53 89

4.0 DEDICATED SHUTDOWN DECAY HEAT REMOVAL SYSTEM . . . 4-1 91

5.0 ALTERNATIVE SELECTION AND INTEGRATION 5-1 99

6.0 ALTERNATIVE IMPACT ANALYSIS 6-1 104
6.1 Methodology and Approach 6-1 104
6.1.1 Objectives 6-1 104
6.1.2 Approach 6-1 104
6.2 Results 6-2 105

7.0 ALTERNATIVE VALUE ANALYSIS 7-1 111
7.1 Core Melt Probabilities 7-1 111
7.1.1 Base Case Probabilities 7-1 111
7.1.2 Alternative Probabilities 7-1 111
7.1.3 Summary of Probabilistic Core Melt Estimates 7-3 113
7.2 Public Risk Estimates 7-3 113
7.2.1 Base Case Estimates 7-3 113
7.2.2 Alternative Estimates 7-13 123
7.2.3 Summary of Probabilistic Risk Estimates 7-13 123
7.3 Non-Quantifiable Values 7-13 123
7.3.1 Reductions in Residual Risk 7-16 126
7.3.2 Effects Upon Equipment Qualification . 7-16 126
7.3.3 Effects Upon Plant Availability 7-17 127
7.3.4 Regulatory Issues 7-17
7.3.5 Summary of Non-Quantifiable Values . . 7-18 128

8.0 UNCERTAINTY AND SENSITIVITY CONSIDERATIONS 8-1 129

8.1 The Nature of the Problem 8-1 129
8.2 The Uncertainty in the Offsite Costs 8-1 129
8.2.1 The Structure of the Offsite Cost Terms 8-2 130
8.2.2 The Uncertainty in Neff 8-2 130
8.2.3 The Uncertainty in C(D) 8-3 131
8.2.4 The Uncertainty in F(D) 8-4 132
8.2.5 The Uncertainty in p(m) and q 8-5 133
8.2.6 Overall Uncertainties in Value Term . . 8-6 134
8.3 The Uncertainty in Combined Onsite and Offsite Costs 8-7 135
8.3.1 The Structure of the Onsite Cost Term 8-7 135
8.3.2 Uncertainty in Neff 8-9 137
8.3.3 Uncertainty in $\Delta p(m)$ 8-9 138
8.3.4 Uncertainty in the Differential Cost of Replacement Power 8-9 137
8.3.5 Uncertainty in the Evaluation of the Loss of the Utility's Investment . . . 8-11 139
8.3.6 Uncertainty in the Cost of Cleanup After Core Melt 8-11 139

139

8.3.7 Uncertainty Due to the Omission of Possible Components on th Onsite Cost . 8-11 139
8.3.8 Summary of the Uncertainty in the Onsite Cost Term 8-12 140
8.3.9 The Uncertainty in the Combined Onsite and Offsite Costs 8-12 140
8.4 Effects of Uncertainties in the Impact Term. . 8-14 142
8.5 The Overall Uncertainty in the Value-Impact Analysis 8-16 144

9.0 INTEGRATED VALUE-IMPACT ANALYSIS 9-1 145
9.1 Methodology 9-1 145
9.1.1 Value and Impact Analysis Variables . . 9-1 145
9.1.2 Value-Impact Analysis Measures 9-4 148
9.2 Results 9-8 152

10.0 SPECIAL ISSUES RELATED TO DECAY HEAT REMOVAL . . . 10-1 160
10.1 Feed and Bleed 10-1 160
10.1.1 Value of Feed and Bleed 10-1 160
10.1.2 Impacts Associated with Feed and Bleed 10-8 167
10.1.3 Operational Issues 10-11 170
10.1.4 Environmental Qualification Concerns 10-11 170
10.1.5 Summary 10-12 171
10.2 Secondary Side Blowdown 10-12 171
10.2.1 Value of Secondary Side Blowdown . . 10-13 172
10.2.2 Systems Required for Successful Depressurization 10-14 173
10.2.3 Summary 10-17 176

11.0 SUMMARY, CONCLUSIONS, AND OBSERVATIONS 11-1 177
11.1 The DHR Assessment 11-1 177
11.1.1 The Internal Analysis 11-1 177
11.1.2 The Special Emergency Analysis . . . 11-1 177
11.1.3 Assumptions and Bases for the Analysis 11-2 178
11.1.4 Base Case Results 11-9 185
11.1.5 The Alternatives and Revised Core Melt Probabilities 11-9 185
11.1.6 The Value-Impact Summary 11-9 185
11.1.7 Discussion of Results 11-9 185
11.2 The Add-on Decay Heat Removal System 11-11 187
11.2.1 Non-Quantifiable Value Issues . . . 11-11 187
11.2.2 Comparison With European Practice. . 11-12 188
11.3 Observations From the Case Study 11-13 189
11.3.1 Introduction 11-13 189
11.3.2 Value-Impact Estimates for Two-Unit Melts 11-14 190
11.4 Observations for the Case Study 11-16 192

REFERENCES

APPENDICES:

A. System Descriptions A-1 198
B. Internal Analysis B-1 235
C. Seismic Assessment C-1 337
D. Fire Analysis D-1 465
E. Internal Flood Analysis E-1 505
F. External Flood Analysis F-1 534
G. Extreme Wind Analysis G-1 589
H. Lightning Analysis H-1 641
I. Sabotage Analysis (under separate cover)
J. Impact Analysis J-1 668
K. Consequence Extrapolation K-1 726
L. Methodology for Value-Impact Analysis L-1 738

LIST OF FIGURES

Figure		Page	
1.1	St. Lucie Site Plot Plan	1-3	3
1.2	Plant Plot Plant	1-4	4
1.3	Reactor Building Elevation 45'	1-5	5
1.4	Reactor Building Elevations 18' and 23'	1-6	6
1.5	Reactor Building Cross Section	1-7	7
1.6	Auxiliary Building at -0.5' Elevation	1-9	9
1.7	Auxiliary Building at 19.5' Elevation	1-10	10
1.8	Auxiliary Building at 43' Elevation	1-11	11
1.9	Fuel Handling Building	1-12	12
1.10	Turbine Building Ground Floor Plan	1-13	13
1.11	Turbine Building Mezzanine Floor Plan	1-14	14
1.12	Turbine Building Operating Floor Plan	1-15	15
1.13	Shutdown Decay Heat Removal Analysis	1-16	16
2.2	Pictorial Representation of the Internal Analysis	2-2	20
3.1	Risk Analysis Flow Chart	3-30	66
3.2	St. Lucie and Surrounding Area (from USGS Topographic Map)	3-32	68
3.3	St. Lucie Plot Plan	3-34	70
4.1	Simplified Schematic for Dedicated Add-on With Emergency Feedwater and High Pressure Make-Up Water	4-2	92
4.2	Potential Add-On SDHR System Fault Tree	4-3	93

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LIST OF TABLES

Table | | Page |
|---|---|---|---|
| 2.1 | St. Lucie Core Melt Accident Sequences . . . . | 2-9 | 27 |
| 2.2 | Accident Sequence Events . . . . | 2-12 | 30 |
| 3.1 | Sequence Core Melt Probabilities for Varying Earthquake Magnitudes - Base Case . . . . | 3-12 | 48 |
| 3.2 | Sequence Core Melt Probabilities for Varying Earthquake Magnitudes - With Modifications . . . . | 3-14 | 50 |
| 3.3 | Vulnerable Structures and Equipment . . . . | 3-36 | 72 |
| 3.4 | Core Melt Frequency Contributors Due to External Flooding at St. Lucie . . . . | 3-40 | 76 |
| 3.5 | Vulnerable Structures and Equipment . . . . | 3-44 | 80 |
| 3.6 | Summary Table of Special Emergency Modifications. . | 3-54 | 90 |
| 4.1 | Add-On System Value Block . . . . | 4-7 | 97 |
| 5.1 | Potential St. Lucie Vulnerabilities . . . . | 5-2 | 100 |
| 5.2 | Initial St. Lucie Insights . . . . | 5-3 | 101 |
| 5.3 | St. Lucie Potential Alternatives . . . . | 5-3 | 101 |
| 5.4 | Preliminary Estimated Core Melt Probabilities Given Modifications Applied to St. Lucie . . . . . | 5-4 | 102 |
| 6.1 | Content of Proposed Alternatives . . . . | 6-4 | 107 |
| 6.2 | Economic Ground Rules and Assumptions from Appendix J . . . . | 6-6 | 109 |
| 6.3 | St. Lucie Impact Analysis Results . . . . | 6-7 | 110 |
| 7.1 | St. Lucie Special Emergency Accident Sequence Probabilities Including Containment Systems . . . . | 7-4 | 114 |
| 7.2 | St. Lucie Accident Sequence to Release Category Mapping . . . . | 7-5 | 115 |
| 7.3 | St. Lucie - Base Case . . . . | 7-7 | 117 |
| 7.3 | St. Lucie - Alternative 1 . . . . | 7-8 | 118 |
| 7.3 | St. Lucie - Alternative 2 . . . . | 7-9 | 119 |
| 7.4 | St. Lucie Dominant Accident Sequences and Special Emergencies Contributing to Release Categories . . . . | 7-10 | 120 |
| 7.5 | St. Lucie Value of Offsite Population Dose in Person-Rem for the Base Case (person-rem/ reactor year) . . . . | 7-12 | 122 |
| 7.6 | St. Lucie Release Category Probabilities (probability per reactor year) . . . . | 7-14 | 124 |
| 7.7 | St. Lucie Expected Value of the Population Dose Within 50 Miles (person-rem per reactor year) . . . | 7-15 | 125 |
| 7.8 | St. Lucie Summary of Values for Averted Offsite Population Dose in Person-Rem per Reactor Year Including Internal and Special Emergencies . . | 7-15 | 125 |
| 8.1 | St. Lucie Nuclear Power Plant - Summary of the Variation in the "Value" Term, if Based on Offsite Costs Only, With the Principal Parameters Subject to Uncertainties . . . . | 8-8 | 136 |

8.2 St. Lucie Nuclear Power Plant - Summary of the Variation in the "Value" Term, if Based on Onsite Costs Only, With the Principal Parameters Subject to Uncertainties 8-13 141
8.3 St. Lucie Nuclear Power Plant - Uncertainty in Combined Offsite and Onsite Costs (present worths in dollars) Due to Factors NOT Dependent on NRC Decisions 8-15 143

9.1 Value and Impact Analysis Impact Variables . . . 9-2 146
9.2 St. Lucie - Value-Impact Analysis Error Factor . 9-5 149
9.3 St. Lucie - Summary of Impacts (based upon 5% Discount) 9-9 153
9.4a St. Lucie - Summary of Values (based on population dose at 50 miles, 5% discount rate) 9-10 154
9.4b St. Lucie - Summary of Values (based on population dose at 50 miles, 5% discount rate) 9-11 155
9.5 St. Lucie - Summary of Value-Impact Analysis (based on population dose at 50 miles, 5% discount rate 9-12 156
9.6 St. Lucie - Summary of Value-Impact Measures (Central Value) 9-15 159

10.1 Contribution by Probability of Core Melt of Accident Sequences Affected by Feed and Bleed . . 10-4 163
10.2 Selected Cut Sets for Sequence T_2MPQL (sequence applies when feed and bleed available) 10-5 164
10.3 Cut Sets for Sequence T_2MPQ LE (sequence applies when feed and bleed available 10-6 165
10.4 Selected Westinghouse Plant Parameters 10-11 170

11.1 Summary of Value-Impact Measures (central value). 11-3 179
11.2 Estimate of Value-Impact Measures for Accidents Involving Both St. Lucie Units 11-8 184

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1.0 INTRODUCTION

1.1 Objectives

The objectives of Task Action Plan A-45 are to evaluate the safety adequacy of decay heat removal (DHR) systems in existing light water reactor nuclear power plants and to assess the value and impact (benefit-cost) of alternative measures for improving the overall reliability of the DHR function if required. To meet these objectives a program was developed which assesses the adequacy of DHR system reliability in existing plants, establishes the feasibility of potential measures to improve DHR, and provides a value/impact analysis of the most promising of such measures.

1.2 Background

Because Unresolved Safety Issue A-45, Decay Heat Removal Requirements encompasses the entire industry, the population of nuclear power plants to be considered for analysis started with approximately 173 units. Seventy units were excluded from study because: 1) they were of special types (e.g., HTGR) or included in the Safety Evaluation Program, 2) there are sufficiently similar units included in the remaining base, or 3) the units were not far enough along in the design/construction process to have information readily available. The remaining 90 plus units formed the base of the TAP A-45 study.

The initial steps in assessing the adequacy of DHR were to: 1) characterize the units in terms of their physical parameters, i.e., number and location of safety pumps, number of redundant emergency power trains, etc., and 2) develop a set of qualitative screening questions against which the plant characteristics could be compared. This set of qualitative screening questions was based upon a thorough review of existing guidance such as the Standard Review Plan and the various Regulatory Guides, previous Probabilistic Risk Assessments, special topical studies such as the Auxiliary Feedwater Studies, etc. The intent was to establish a set of questions which would reveal potential deficiencies in DHR capabilities for both Design Basis Events and for beyond Design Basis Situations. After review and reduction, some 135 questions remained for use in the qualitative screening process. There are four possible outcomes of such a qualitative screening: the plant meets or does not meet the postulated condition; there is insufficient plant information in the data base to evaluate the design; or the question does not apply to that plant. It must be remembered that the sole purpose of this qualitative screening was to highlight potential DHR vulnerabilities for further study. Because each plant has a unique configuration, an inability to meet the conditions of a particular question at one plant may be more or less serious than a similar inability at another plant. For example, consider the case of a single valve in a DHR water supply line. There is a screening question designed to identify such valves which, if closed, would fail the system. If this valve is in the line from the only source of water, it may be a potentially serious single failure mode. However, if there exist redundant sources of water, redundant lines, or other systems which perform the same function, the single valve may be of little concern. Such differences can only be established by detailed analysis. Nevertheless, the qualitative screening serves as a tool to focus program resources and direction.

On the basis of the initial qualitative screening, approximately twenty plants, which included all vendor types, were potential candidates for further study. From this group seven plants were selected. This selection process was subjective and took into account other issues in which a particular plant might be involved, operational status, utility willingness to participate, and similar concerns. The St. Lucie Nuclear Plant was one of those selected for more detailed study.

1.3 The St. Lucie Nuclear Plant

St. Lucie Units 1 and 2 are Combustion Engineering closed-cycle pressurized light water steam supply systems of the two-loop class. Unit 1 has a rated core heat output of 2560 MW_t and an electrical output of approximately 827 MW_e. The plant is located on Hutchinson Island, Florida, with the turbine building oriented parallel to State Road A1A and the shoreline of the Atlantic Ocean, and the reactor containment structure located on the east, or seaward, side of the turbine building. The reactor auxiliary building is located perpendicular and close to the turbine building, oriented in an east-west direction. The fuel handling building is located next to the reactor containment building and the reactor auxiliary building, oriented in a north-south direction. The service building is located north of the turbine building.

Refer to the Site Plot Plan, Figure 1.1, and the Plant Plot Plan, Figure 1.2, for the site general layout.

The containment structure houses the nuclear steam supply system (NSSS), consisting of the reactor, steam generators, reactor coolant pumps, pressurizer, and some of the other reactor auxiliaries. The containment structure is served by a polar bridge crane. Figure 1.3 shows the reactor building plan at elevation 45' with only the main components represented. Figure 1.4 shows the 18' and 23' elevations with more detail, while Figure 1.5 gives a sectional view of the containment.

The auxiliary building houses the waste management facilities, engineered safety features components, heating and ventilating

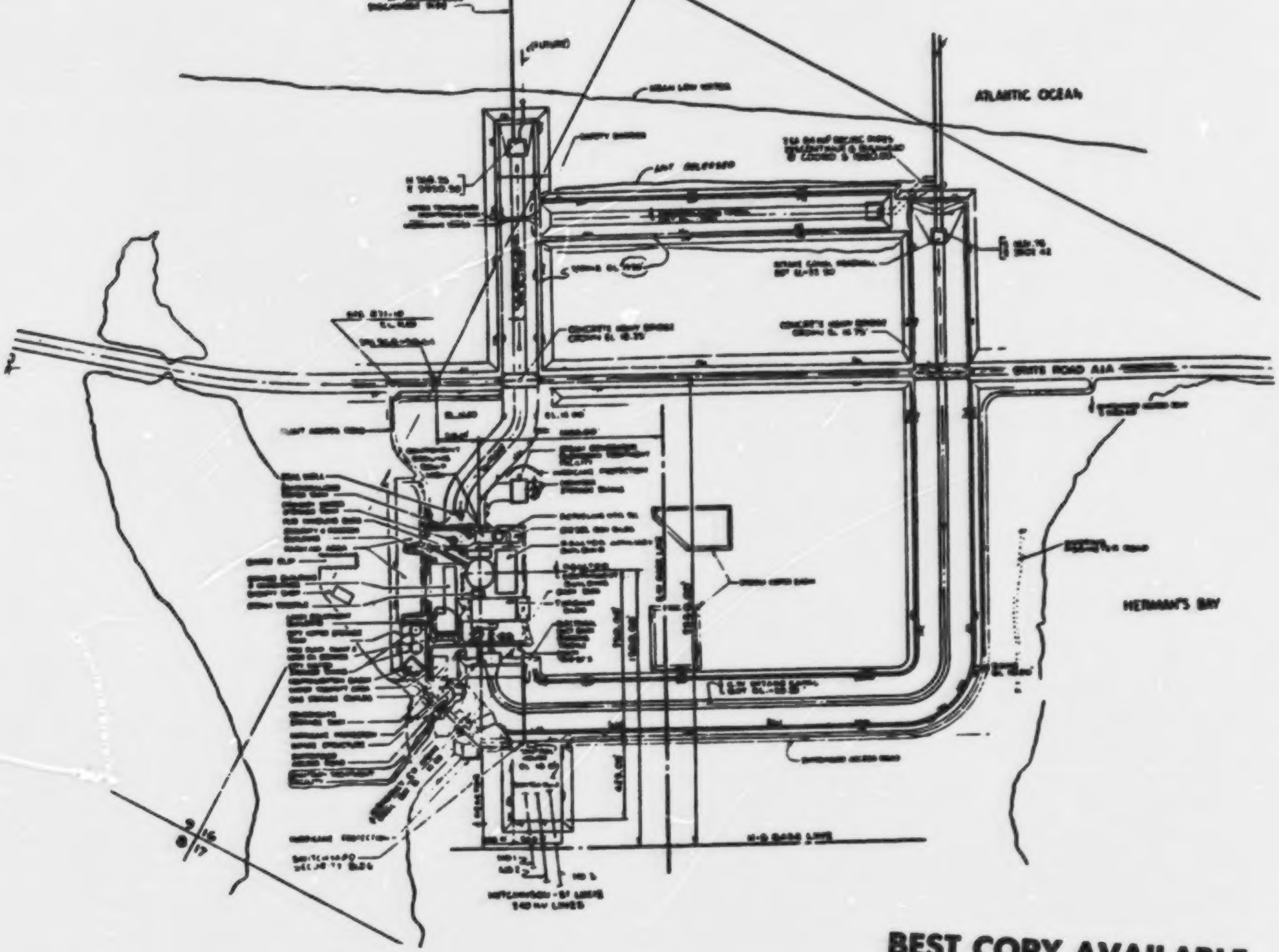
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Figure 1.1 St. Lucie Site Plot Plan

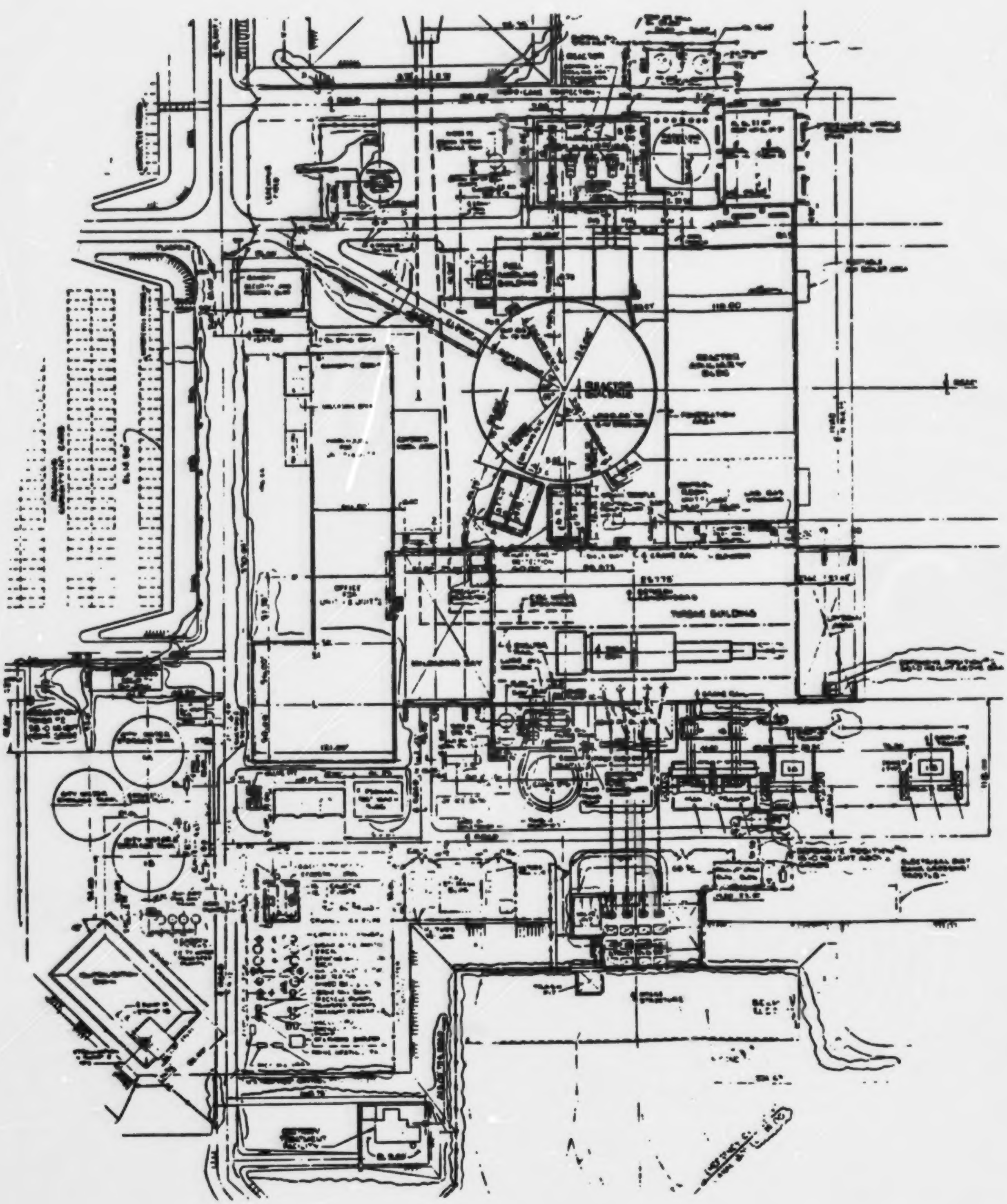
REACTOR BUILDING
REACTOR AUXILIARY BLDG
TURBINE BUILDING

Figure 1.2 Plant Plot Plan

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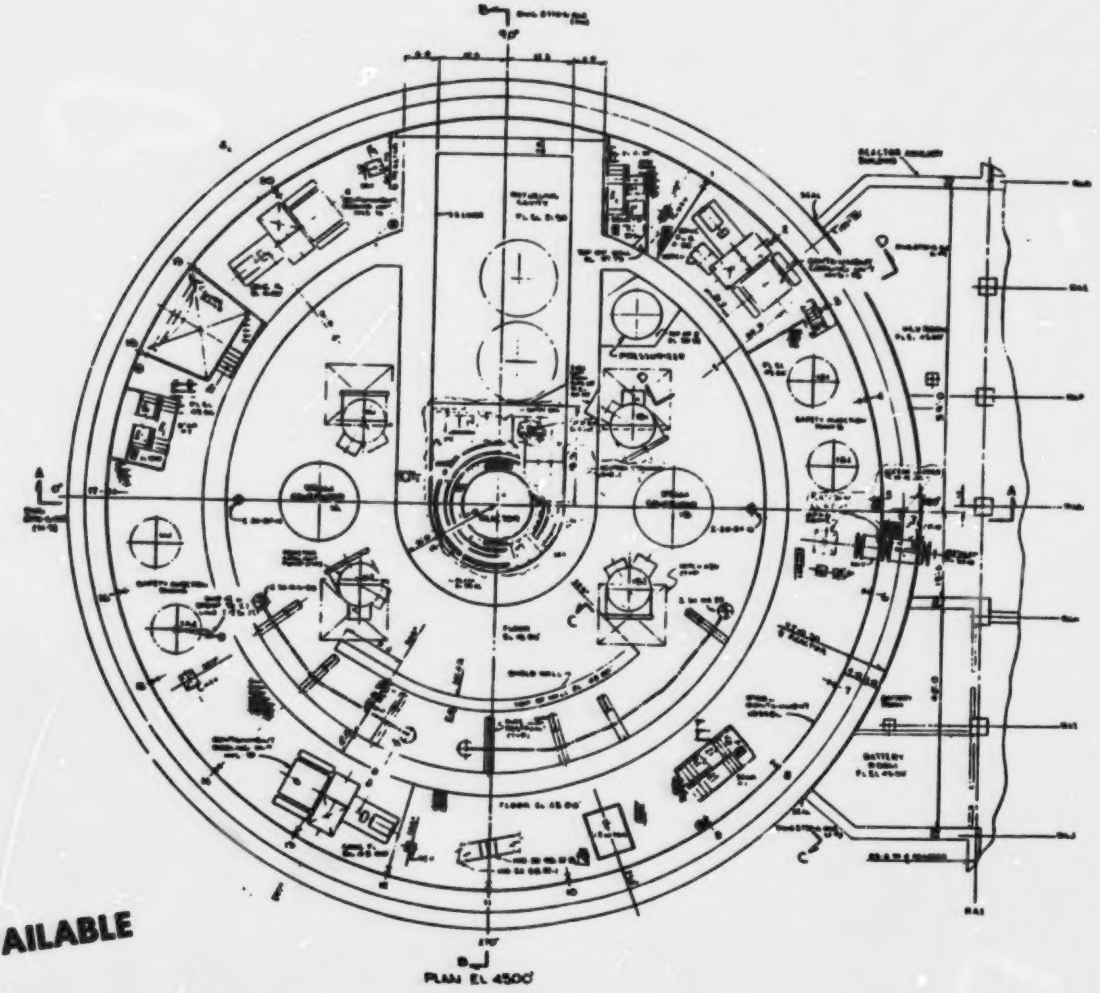
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Figure 1.3 Reactor Building Elevation 45'

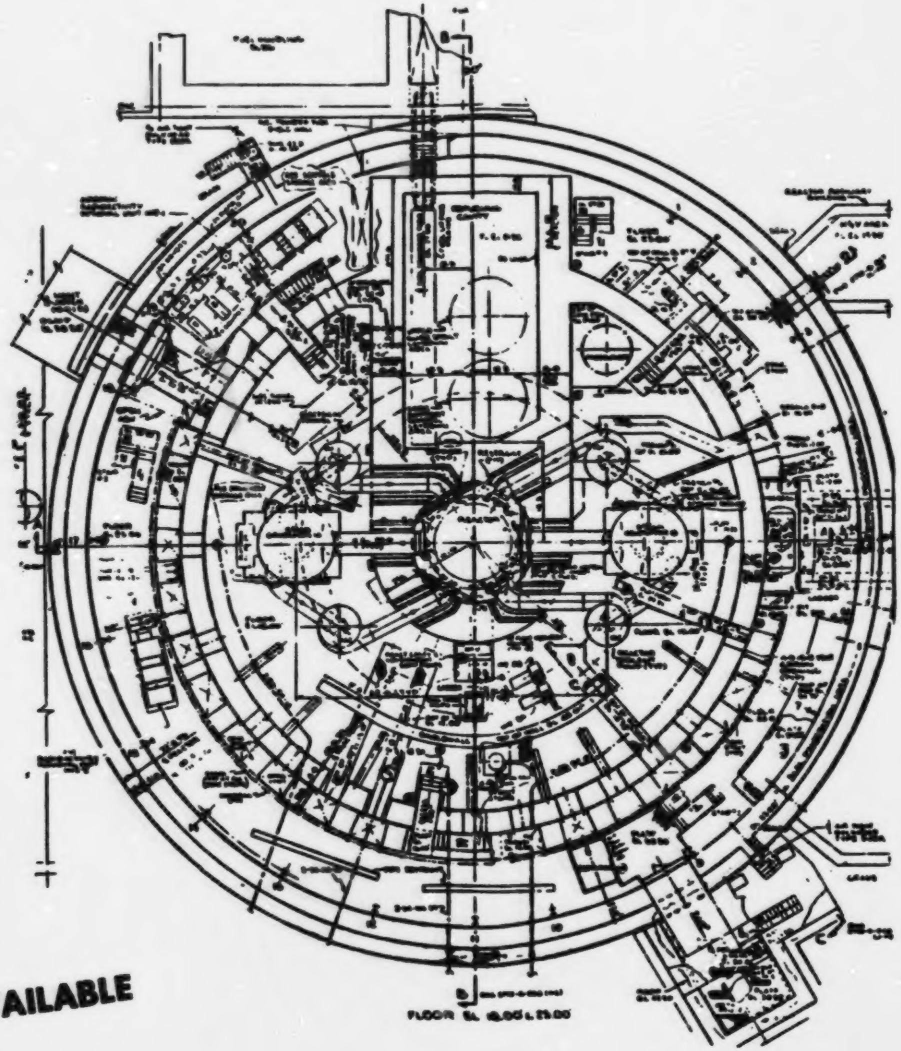
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Figure 1.4 Reactor Building Elevations 13' and 23'

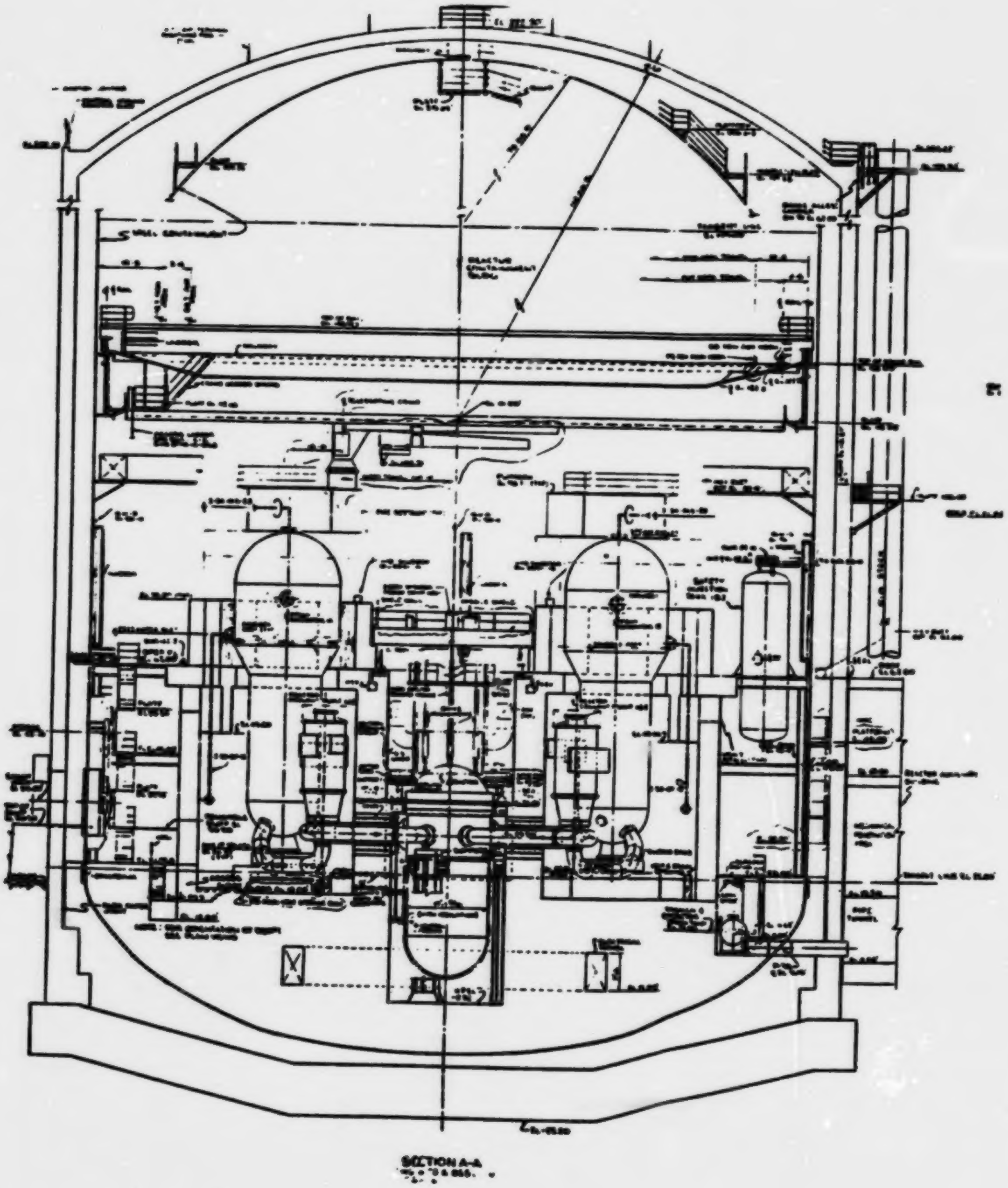

Figure 1.5 Reactor Building Cross Section

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system components, switchgear, laboratories, offices, laundry and control room. Figure 1.6 illustrates the building layout at -0.5', Figure 1.7 shows the 19.5' elevation, and Figure 1.8 gives the plan at the 43' elevation.

The fuel handling building contains the spent fuel pool and new fuel storage facilities, as well as the cooling and purification equipment for the fuel pool. The fuel is transferred from the reactor containment building to the fuel handling building through the fuel transfer tube. Figure 1.9 shows three elevation views of the fuel handling building layout.

The turbine building houses the turbine generator, condensers, feedwater heaters, condensate and feedwater pumps, turbine auxiliaries and certain of the switchgear assemblies. Figure 1.10 shows the turbine building ground floor plan (elev. 19.5'), Figure 1.11 shows the mezzanine floor plan (elev. 39.5'), and Figure 1.12 shows the operating floor plan (elev. 62').

The service building provides offices, shops and warehouse space, and is located next to the turbine building unloading bay.

These plant features are discussed in varying detail in the various special emergency analyses. Additional details may also be found in Appendix I.

1.4 The Shutdown Decay Heat Removal Analysis

As noted above, the St. Lucie plant was identified in the initial qualitative screening as having sufficient potential vulnerabilities to warrant additional study. This initial screening and then subsequent identification and evaluation of alternatives for improving DHR are the subjects of this report. Some potential accidents which could result in core melt have not been analyzed. Since the purpose of this program is to study the adequacy of shutdown decay heat removal systems, large LOCAs, reactor vessel ruptures, interfacing system LOCAs, and anticipated transients without scram (ATWS) are excluded. Furthermore, special issues being studied in depth elsewhere, such as pressurized thermal shock, are also excluded.

The flow of the analysis and value-impact assessment are illustrated in Figure 1.13. The internal analysis proceeds along the well-documented approach used for other probabilistic risk assessments (PRA). The potential accident initiating events are identified and combined with required functions in event trees. Then using Boolean solutions to the fault trees describing the safety systems required to perform the safety functions combined with the initiating events, accident sequences are defined which can lead to situations resulting either in safe conditions or core melt. The probability of those sequences occurring which result in core melt was

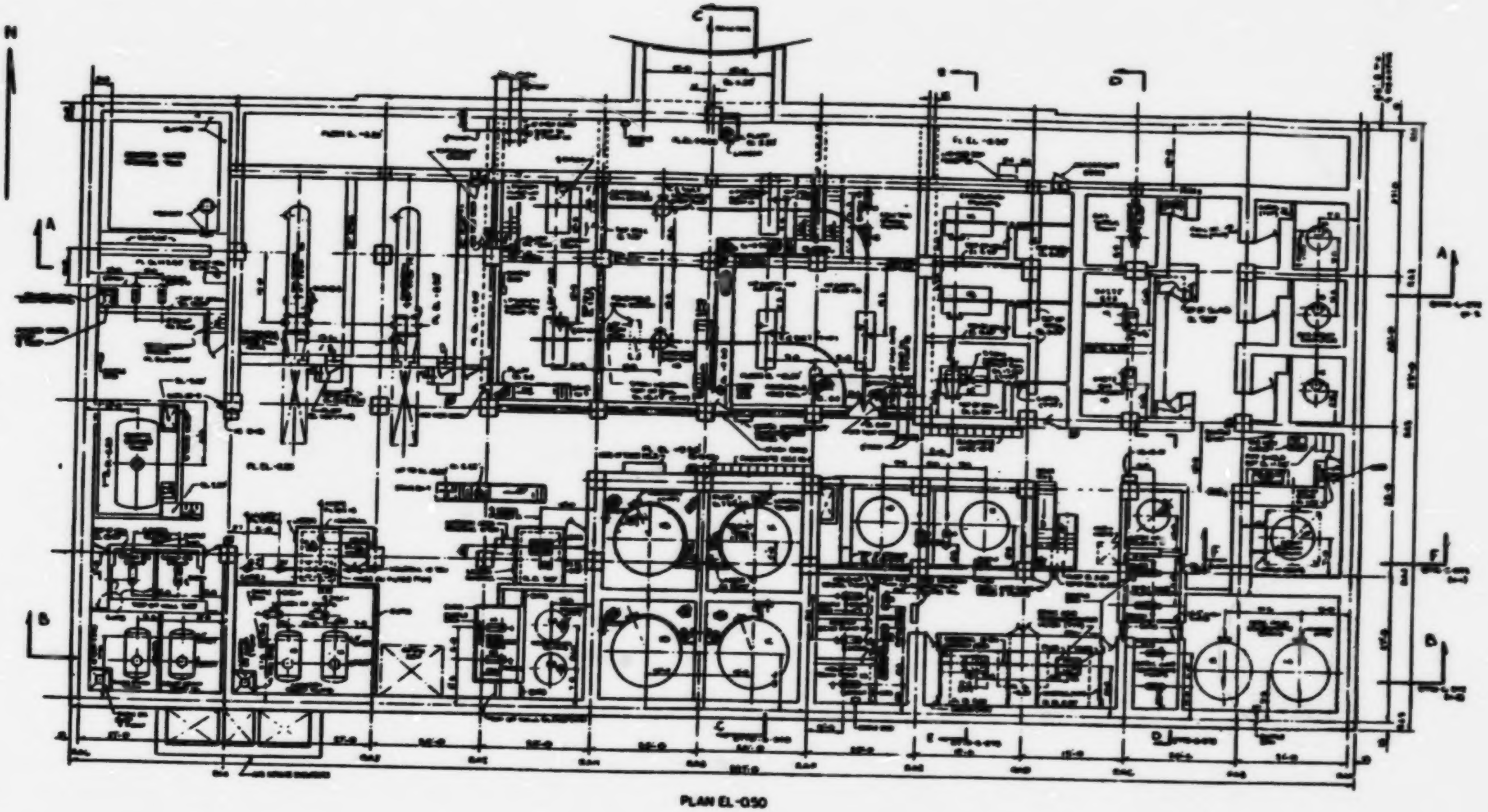
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Figure 1.6 Auxiliary Building at -0.5' Elevation

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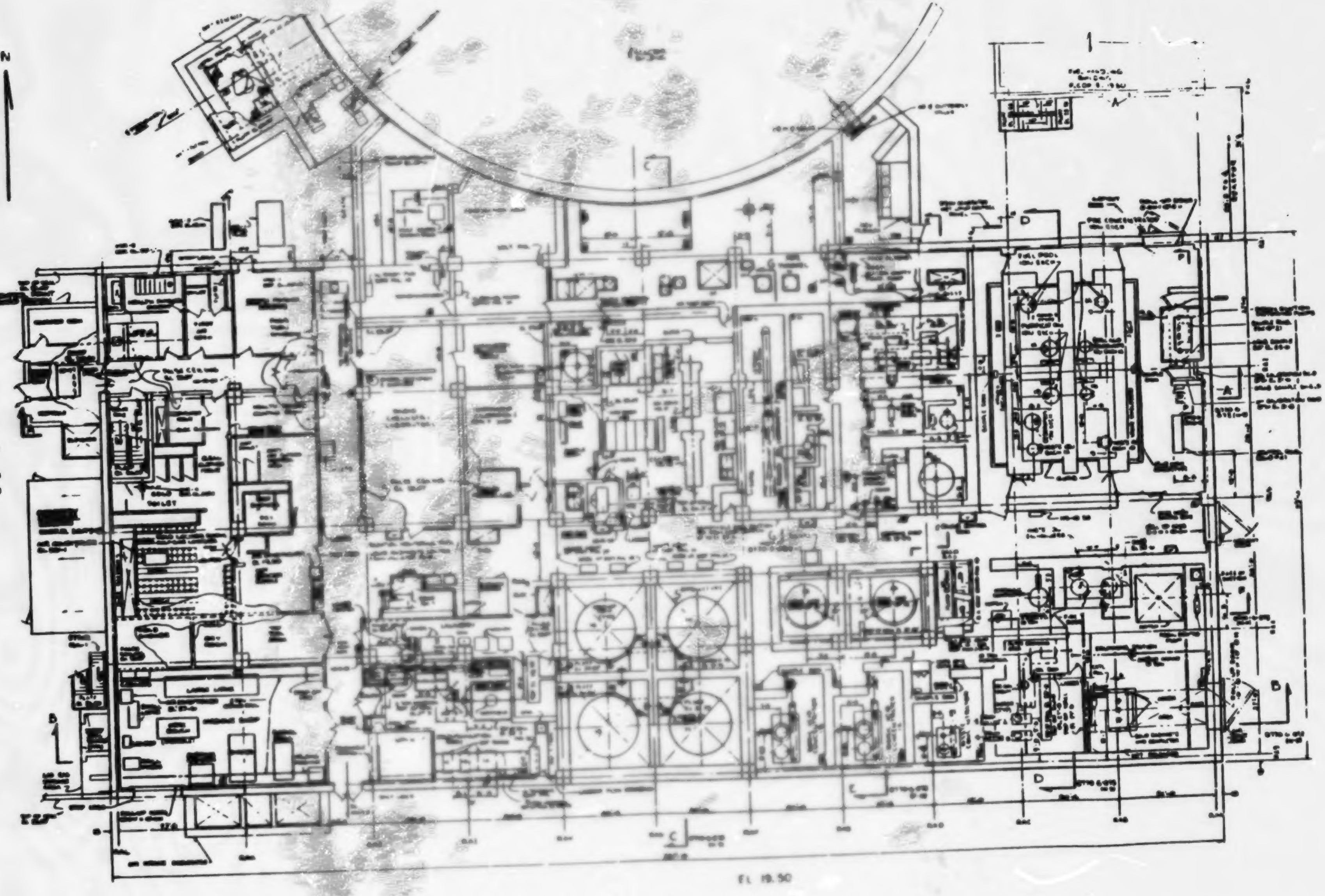

Figure 1.7 Auxiliary Building at 19.5' Elevation

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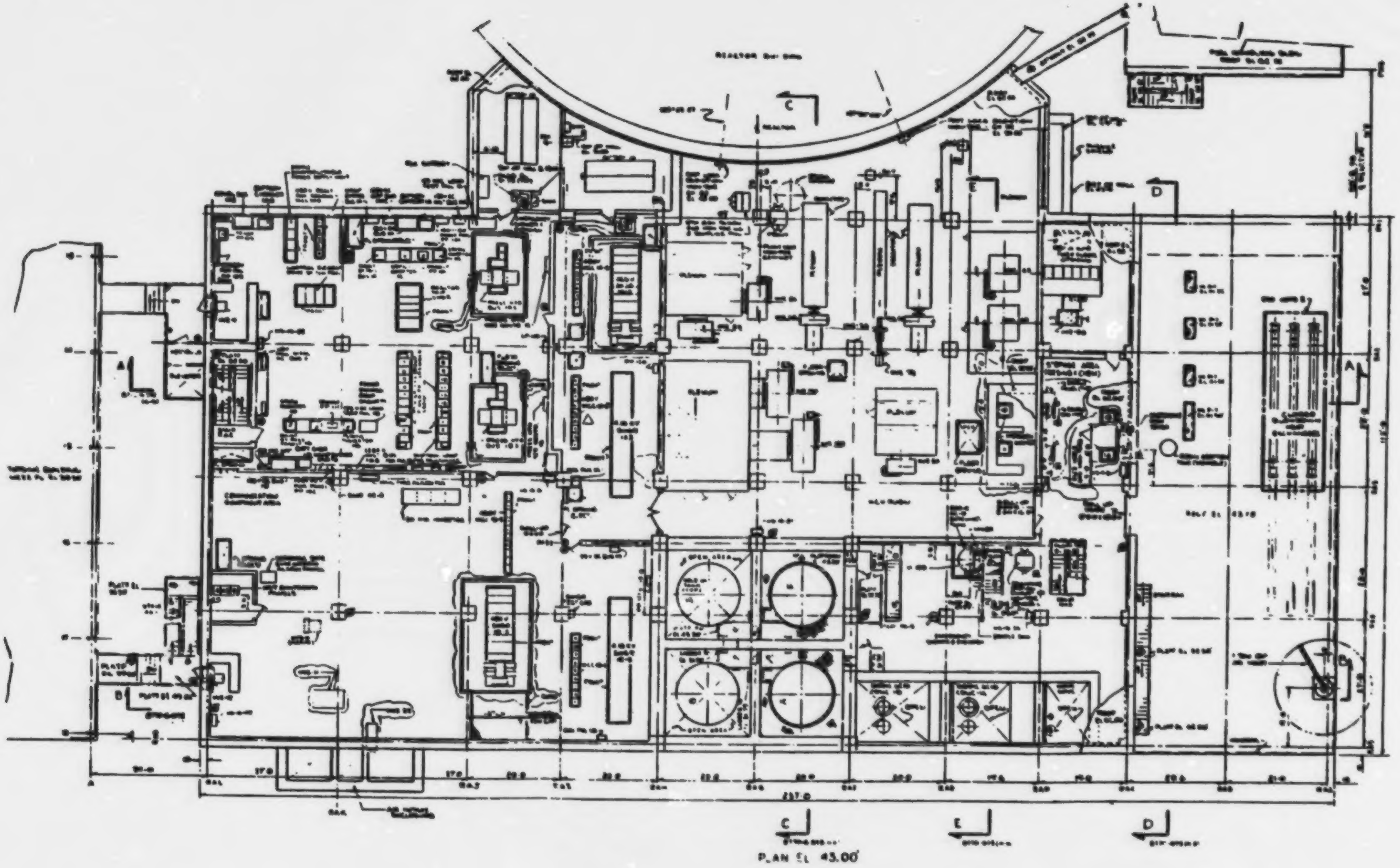
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Figure 1.8 Auxiliary Building at 43' Elevation

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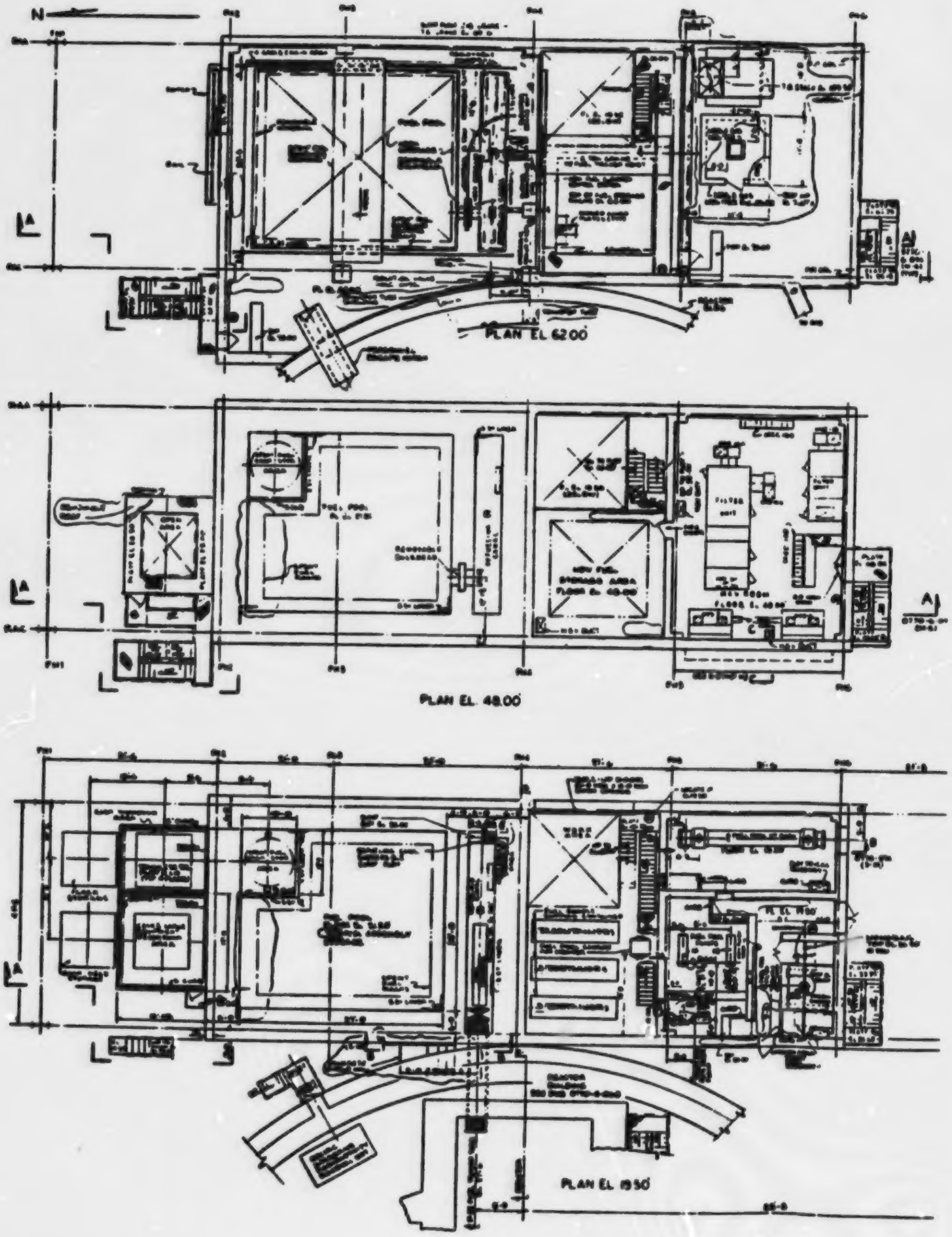
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Figure 1.9 Fuel Handling Building

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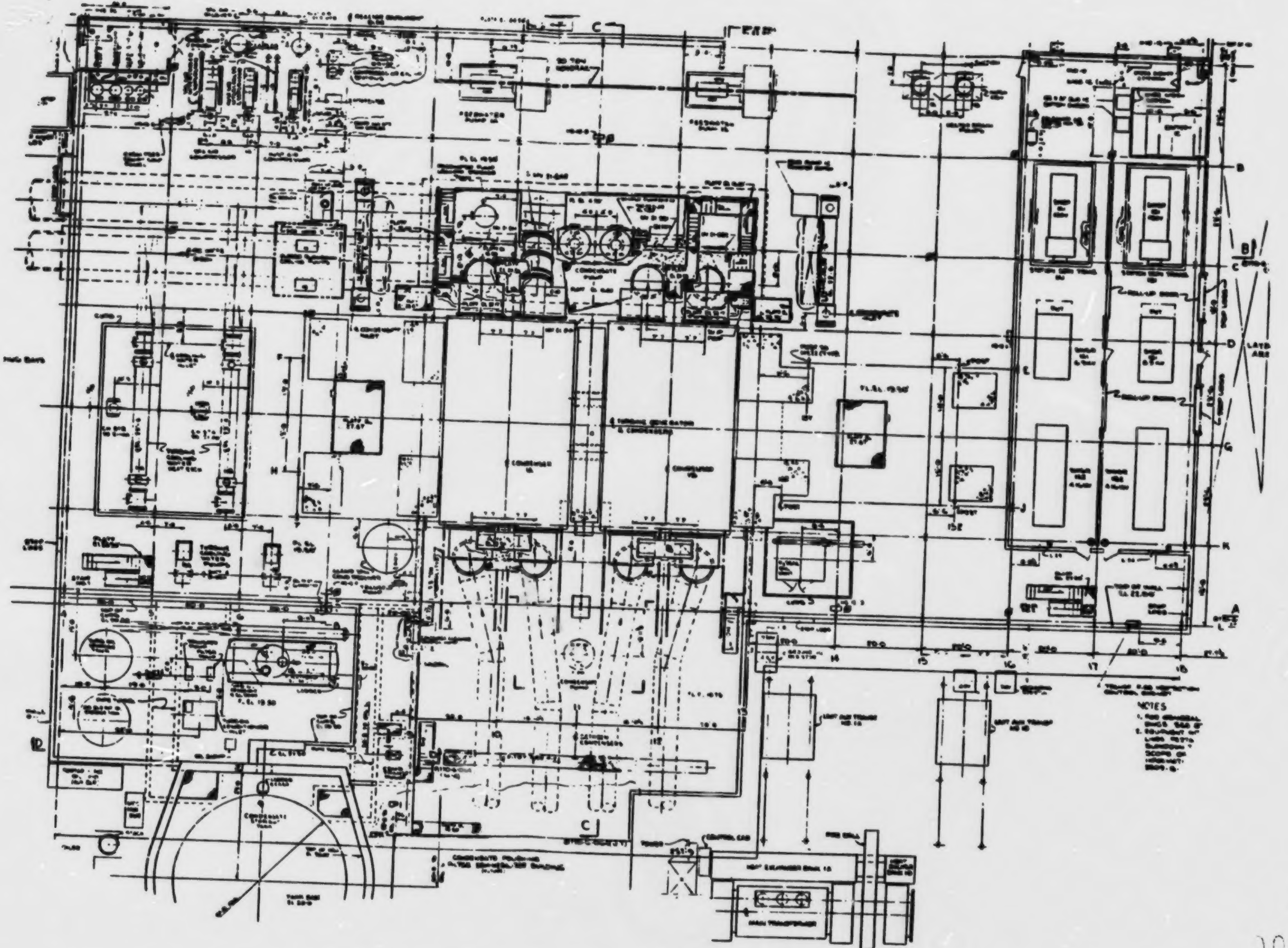
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Figure 1.10 Turbine Building Ground Floor Plan

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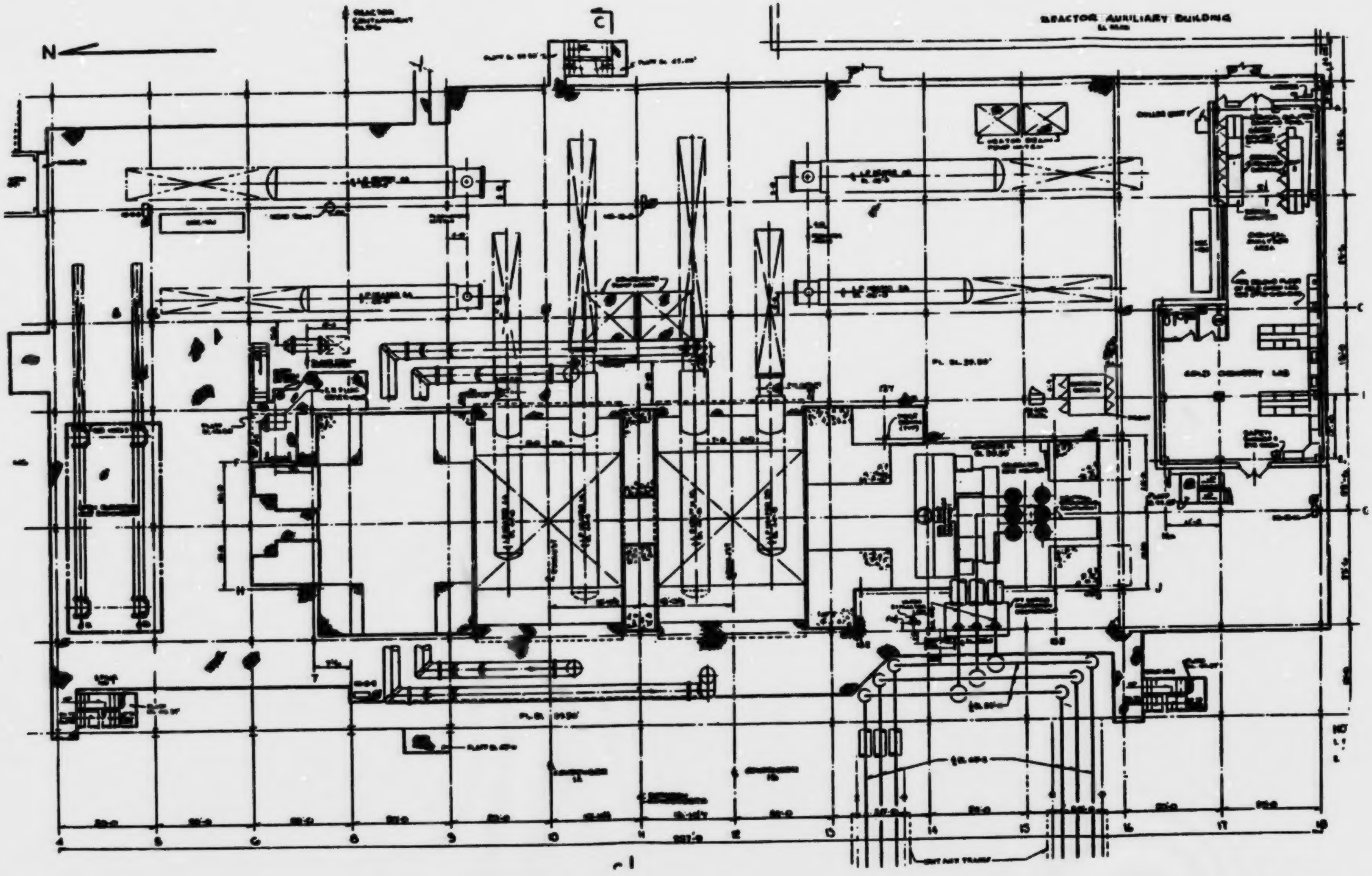
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Figure 1.11 Turbine Building Mezzanine Floor Plan

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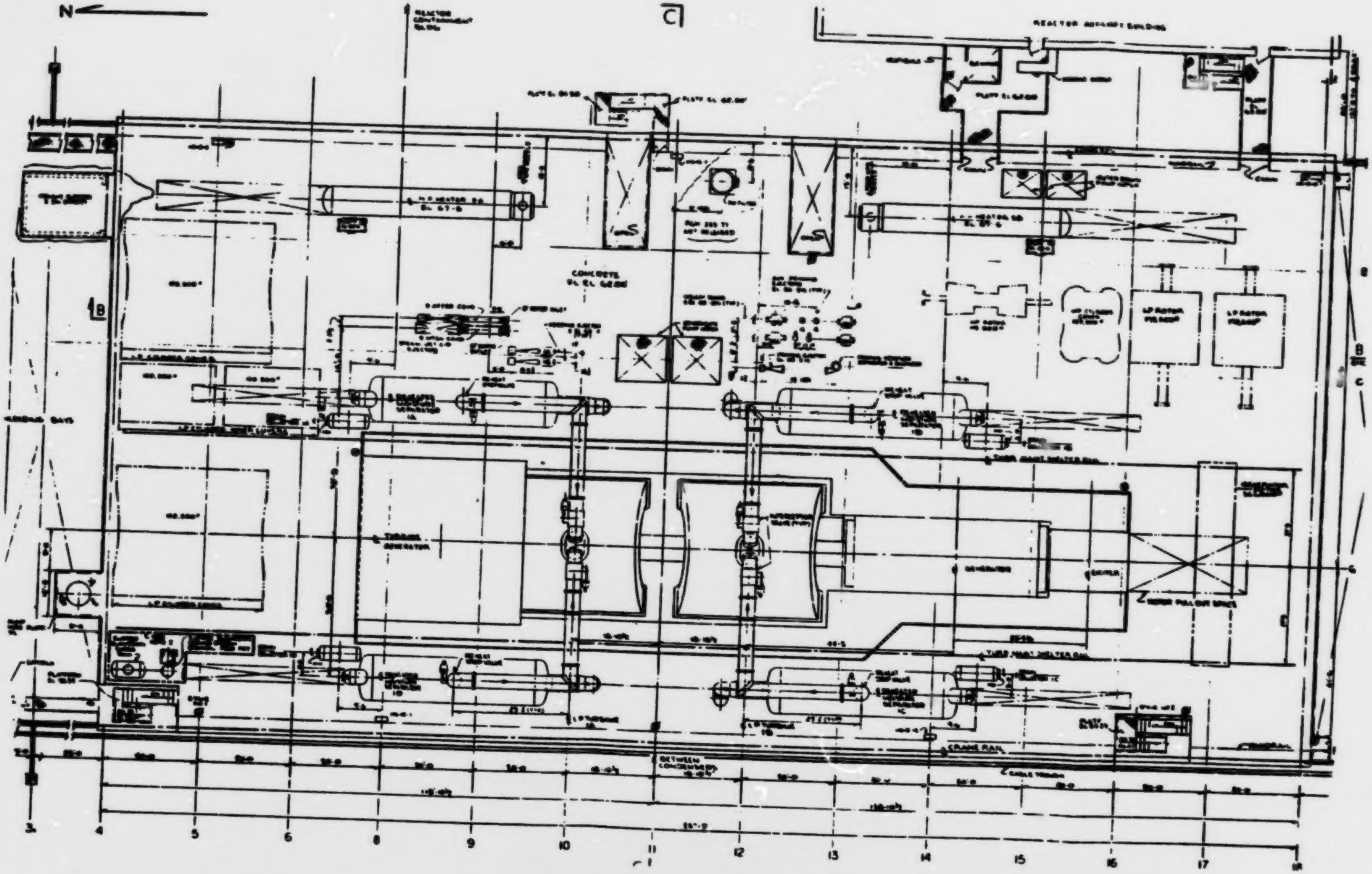
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REACTOR AUXILIARY BUILDING
CONCRETE
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GENERATOR
CRANE RAIL
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Figure 1.12 Turbine Building Operating Floor Plan

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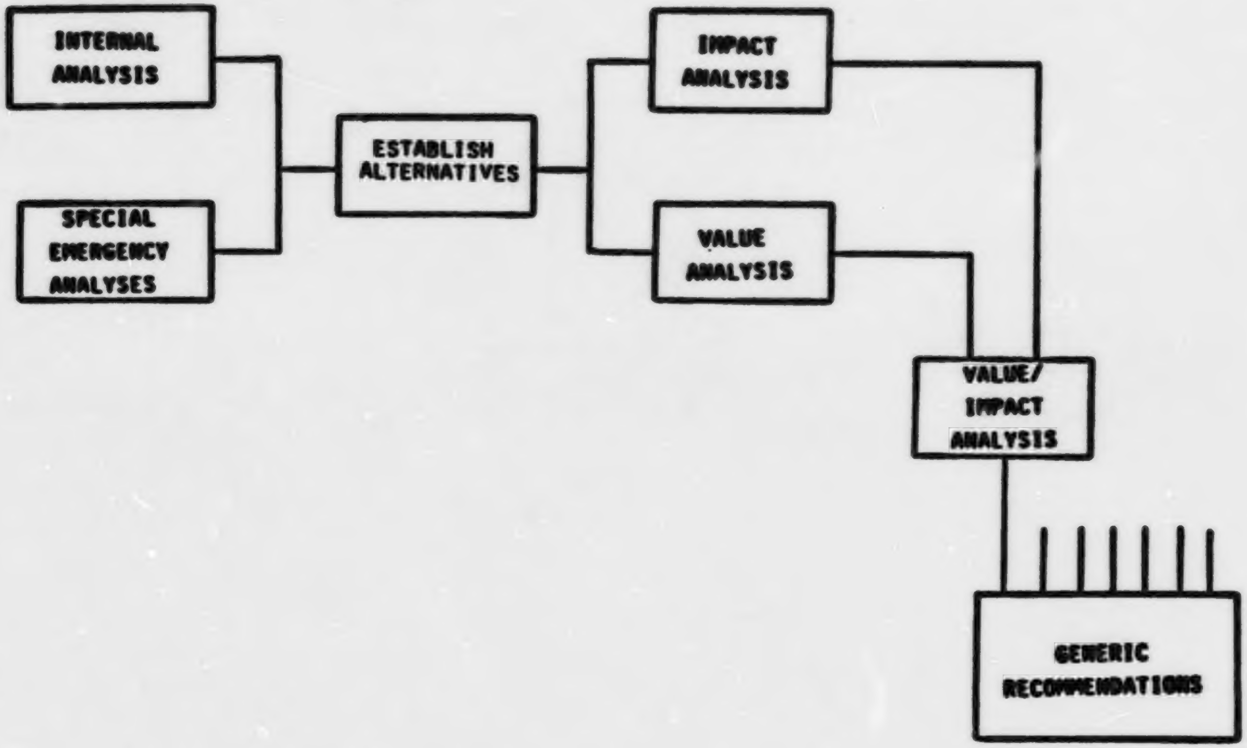
INTERNAL ANALYSIS
SPECIAL EMERGENCY ANALYSES
ESTABLISH ALTERNATIVES
IMPACT ANALYSIS
VALUE ANALYSIS
VALUE/ IMPACT ANALYSIS
GENERIC RECOMMENDATIONS

Figure 1.13 Shutdown Decay Heat Removal Analysis

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quantified using available computational tools. This quantification also identifies the specific random failure events (e.g., equipment failure, operator error) which contribute to the core melt and the magnitude of that contribution. In addition, for core melt sequences the performance of containment systems was also examined. Then using the results of past PRAs, these results are used to determine the containment failure modes. This leads to delineation of potential release category probabilities which can be used to establish public risk in terms of expected population radiation dose and other measures. The results of the internal analysis are estimates of the probability of specific failures and their contribution to core melt and public risk. This information can then be used to suggest modifications to equipment or procedures which reduce or eliminate those failures. This process is discussed further in Section 2 and Appendix B.

The special emergency analyses proceed on a parallel and similar path although perhaps more qualitative than the internal analysis and certainly more dependent upon engineering judgment. Generally the special analysis for earthquake, flood, and high winds proceed by identifying the hazard and its frequency of occurrence. Then an assessment of the response of the plant to this hazard is established. A key aspect of this assessment is an onsite inspection of the plant and its equipment. Once the response is established, individual equipment fragilities are established. Then using the internal event fault trees and events, the potential contribution of specific failures to core melt probability were estimated. Using this information modifications were defined again to eliminate or reduce the effects of certain vulnerabilities.

The special emergency analyses for fire and internal flood proceed in a slightly different fashion. First, potentially significant fire areas are identified using the transient event trees and system fault trees to establish the critical front line and support systems. Second, the physical arrangements of equipment and potential fire sources are verified by a site visit. The effects of the special emergency are then quantified based on historical fire occurrence frequencies, analytical models of fire growth, and fire suppression probabilities. In addition, random failures and human factors are considered as applicable. Based upon these results, modifications designed to reduce core melt can be proposed.

The special emergency analyses are discussed in Section 3 and Appendices C through I.

Based upon initial estimates of the possible reductions in core melt probability that might be achieved by the various modifications, the modifications are combined into groups for engineering evaluation of feasibility and impact. These combinations are called alternatives. The alternatives are reviewed by the Architectural Engineer (AE) and the plant staff for feasibility and reasonableness. Based upon this initial review, the AE proceeds with conceptual design. After an onsite inspection to insure that the proposed designs could be implemented and to gather site specific cost and related data, the AE develops the impact data. These data include, but are not limited to, capital costs for engineering and installation, radiation exposures incurred, maintenance and operational costs. Once the conceptual designs are reasonably well established, the internal and special emergency analyses are rerun for each alternative to establish their value in terms of reduced core melt probability and reduced public risk. These analyses are described in Sections 6 and 7 respectively. The impacts and values are then integrated into a value-impact analysis. This analysis is structured according to existing NRC guidelines. Further discussion of this analysis is provided in Section 9. A brief uncertainty analysis is provided in Section 8 and an analysis of special issues (i.e., bleed and feed, and secondary blowdown mode of operation) is provided in Section 10. Finally, the analysis suggests those generic insights or conclusions that might be inferred from the results for one plant (Section 11).

2.0 INTERNAL ANALYSIS - METHODS AND RESULTS

2.1 Analysis Methods

The internal analysis methods are described in this section using the pictorial representation of the internal analysis shown in Figure 2.1 as a guide. After the various tasks are explained an example will be given in the symbols that are typical of probabilistic risk assessments and of this analysis for St. Lucie.

2.1.1 Modeling

The process begins with the selection of the initiating events to be considered. Since the purpose of this program is to study the shutdown decay heat removal (SDHR) function adequacy, large LOCAs, reactor vessel ruptures, interfacing system LOCAs, and anticipated transients without scram (ATWS) were excluded. Furthermore, special issues being studied elsewhere such as pressurized thermal shock were also excluded. Thus, the initiating events to be analyzed are:

Small LOCA
Loss of Offsite Power Transients
Transients resulting from initial Loss of the Power Conversion System (PCS)
Transients with offsite power and the PCS initially available
Transients resulting from the loss of an AC or DC bus

In order to develop the event trees for each of these initiating events, the success criteria for the systems contributing to the SDHR function were delineated using the plant Final Safety Analysis Report and discussions with plant personnel as the sources of information. The objective was to define the success criteria and potential options to address the various accident conditions realistically and not be overly conservative. The front line systems that were considered are:

Auxiliary Feedwater System (L)
High Pressure Injection (D_1) & Recirculation System (H_1)
Low Pressure Injection (D_2) & Recirculation System (H_2)
Power Conversion System (M)
Pressurizer Safety and Relief Valves (P)
Secondary Safety and Relief Valves (Q)

In addition, the support systems considered are:

Emergency AC and DC Power Systems
Service Water System
Component Cooling Water System
Emergency Safeguards Actuation System

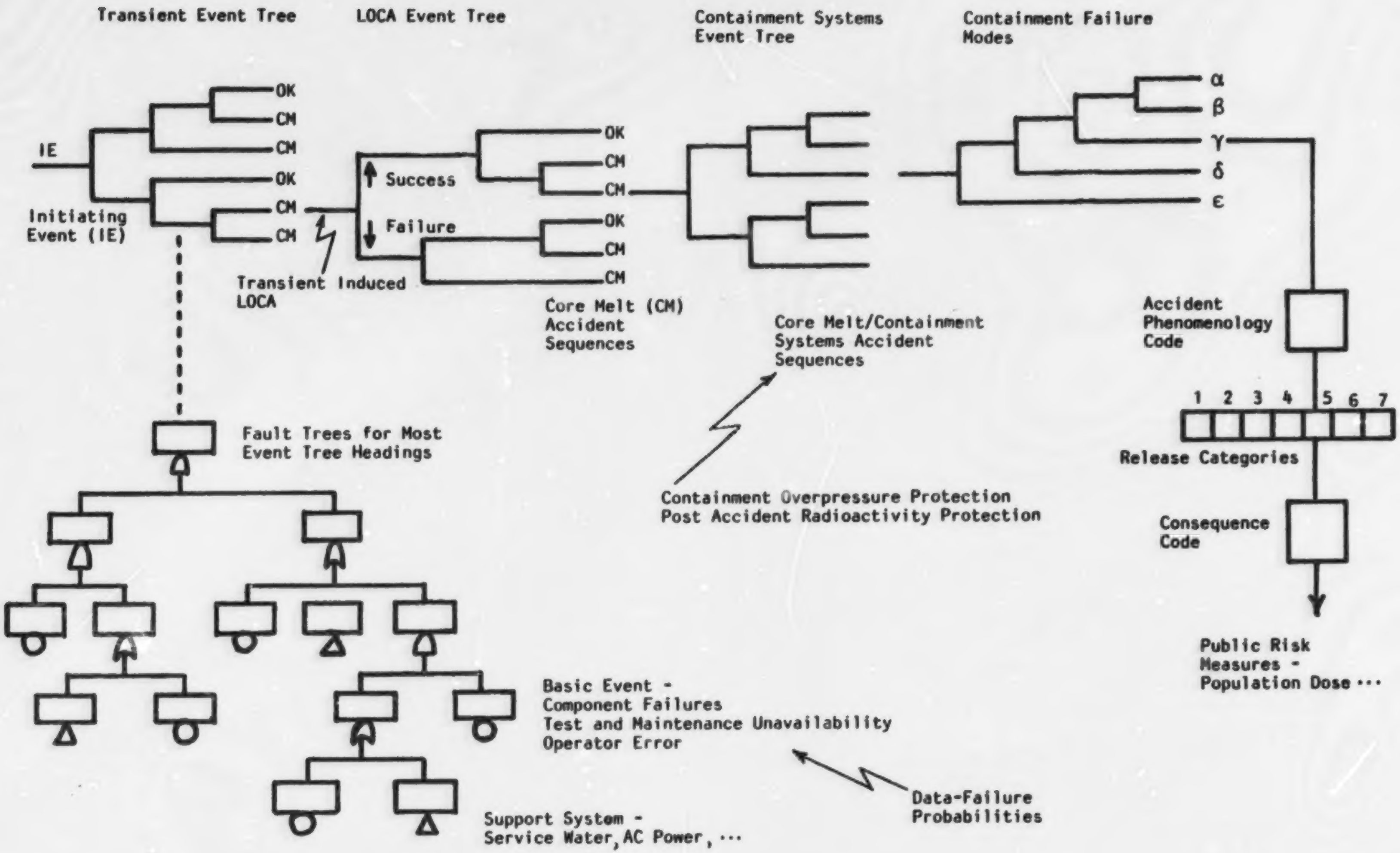
Transient Event Tree
LOCA Event Tree
Containment Systems Event Tree
Containment Failure Modes
IE
OK
CM
CM
OK
CM
CM
Initiating Event (IE)
Success
Failure
OK
CM
CM
OK
CM
CM
Transient Induced LOCA
Core Melt (CM) Accident Sequences
α
β
γ
δ
ε
Core Melt/Containment Systems Accident Sequences
Accident Phenomenology Code
1
2
3
4
5
6
7
Release Categories
Fault Trees for Most Event Tree Headings
Containment Overpressure Protection
Post Accident Radioactivity Protection
Consequence Code
Public Risk Measures -
Population Dose ···
Basic Event -
Component Failures
Test and Maintenance Unavailability
Operator Error
Data-Failure Probabilities
Support System -
Service Water, AC Power, ···

Figure 2.1 Pictorial Representation of the Internal Analysis

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Each of the front line systems or combinations of those systems, e.g., bleed and feed mode (E) using the high pressure system and pressurizer safety and power operated relief valves, becomes a column heading or decision branch in the event trees. The event tree starts with an initiating event and successively asks whether or not the systems (in each of the headings) are successful or fail. In some branches previous success/failure answers make later decisions irrelevant resulting in an asymmetric event tree that most simply represents the sequence of events leading to the potential outcomes, success or core melt. A transient event tree is shown in Figure 2.1 in the upper left hand part of the diagram. A LOCA event tree is shown to the right of the transient event tree representation. LOCAs may be initiated independently or can result from transient induced LOCAs as depicted in the diagram. The final result in any of the cases, transient, transient induced LOCA, or LOCA, is an accident sequence made up of the success or failure of events (i.e., event tree headings representing front line systems) leading to either success or core melt.

The accident sequences will be carried further into the analysis, but first it is important to look at the development of the event tree headings. These are modeled on fault trees such as the one shown in the lower left side of Figure 2.1. Each fault tree is composed of all the components in the flow paths of the systems involved in an event tree heading. Often the event tree heading is simply one specific system. Sometimes more than one system may be combined in a fault tree. In any case the fault tree represents all the potential failure combinations that could lead to failure of the system based on its system success criteria, e.g., flow from 2 out of 3 pump trains. The component is the key to the development of the fault tree. At the component level all the ways that component can fail are modeled including the support systems required by that component. Interactions between components and trains of a system are implicitly modeled by the step-by-step process of constructing the fault tree and identifying all the potential failures.

In this program the event trees are small and the fault tree medium in size which reflects the detail permitted by the scope of the program. This seemed to be the proper mix to meet the objectives of the program.

The outcome of each branch of an event tree is an accident sequence such as $S_2ML\bar{D}_1P_1$, where the initiating event is a small LOCA followed by loss of the PCS, failure of the auxiliary feedwater system, success of the high pressure injection system, and finally, failure of the pressurizer power operated relief valves leading to an early core melt.

The last part of the modeling task is to establish the data base to be used to quantify the accident sequences. Each initiating event frequency and basic fault tree event probability must be derived from data or engineering judgment. The data used in this analysis is essentially generic in nature. Within the analysis some failure rates apply to many components. For example, the local fault (LF) of a motor operated valve (V) which is normally closed (NC) and fails to open, i.e., fails closed (FC), VCC-LF is 3E-3 per demand. Thus CCW30-VCC-LF, CCW15-VCC-LF, AFW18-VCC-LF, etc., would all be assigned the same failure probability.

The model then consists of event trees for each initiating event, fault trees for each event used in the event trees, and the data base (or value block in the computer program terminology).

2.1.2 Accident Sequence Analysis

The output of the SDHR model is a set of accident sequences in which each accident sequence is composed of the product of the Boolean equation for each of the accident sequence events. In the St. Lucie analysis there were 64 distinct accident sequences identified potentially leading to core melt since bleed and feed and secondary blowdown are available modes of operation. The SETS Boolean reduction computer code[6] was run on each of these accident sequences to obtain the core melt probability and the significant combinations of basic events that result in core melt. The Boolean reduction results in a sum of products of basic events. Each of these products is called a cut set such as PUMP1 and VALVE7 and OPERATOR ACTION A.

Every accident sequence is made up of unmodeled and modeled events. For example, in the accident sequence $S_2ML\bar{D}_1P_1$, the S_2M events are not modeled but are assigned a frequency of S_2 and a probability for M from the data base. The events $L\bar{D}_1P_1$ are the Boolean part of the accident sequence leading to the cut sets. The product of $f(S_2)*p(M)*p(L\bar{D}_1P)$ is the core melt contribution for this accident sequence. The sum of the core melt probabilities for all accident sequences is the SDHR core melt probability. However, there are recovery actions that can be taken which were not included in the basic model due to the complexity that would have been incurred. There are also scenarios which represent test and maintenance unavailabilities of redundant safety trains which would not be allowed by technical specifications. Therefore, the next step is to apply the appropriate recovery factors to all the significant cut sets of the dominant accident sequences (that is the accident sequences that contribute the most to core melt ~ 90+%) and eliminate the cut sets which represent Tech Spec violations. There were 12 accident sequences in the St. Lucie analysis carried through this "recovery" analysis. Application of recovery results in a lower SDHR core melt probability.

2.1.3 Vulnerability Identification

The most significant cut sets of the dominant accident sequences after recovery has been applied are the basis for identifying the potential internal vulnerabilities. When these cut sets are examined and grouped by types of failures a pattern develops in that certain basic event failures appear in the cut sets from several accident sequences. The outcome is a ranking of potential vulnerabilities, the cause of each vulnerability and the contribution of each vulnerability to the total SDHR core melt probability. The objective is to identify those vulnerabilities which contribute something over 80% of the total SDHR core melt probability and then address possible modifications to reduce or eliminate them.

A further screening of the accident sequences is provided by the vulnerability identification resulting in only 5 accident sequences requiring further consideration. While the vulnerabilities were selected based on core melt probability, the input needed for the value-impact analysis is risk to the public. The risk measure chosen is population dose in person-rem. In order to obtain population dose the dominant accident sequences must be extended to include containment failure, environmental transport, and consequence analyses.

2.1.4 Containment System Integration

In early PRAs it was assumed that failure of the containment resulted in sump flashing and subsequent failure of the emergency core cooling systems due to pump cavitation. Based upon discussions with several analysts and the St. Lucie staff, this is an overly conservative approach, thus in this analysis core melt does not depend on containment systems. The containment systems event tree for St. Lucie is depicted in Figure 2.1 on the extension of one of the core melt (CM) accident sequences. Each of the 12 dominant accident sequences was rerun with each of the 6 Boolean containment systems event tree outcomes to establish the probabilities for the containment systems outcomes or states. This is the input for the next stage of the analysis.

The containment systems considered are:

Containment Spray Injection (C) and Recirculation (F) System
Containment Air Recirculation Cooling System (Y)

Outcomes of the containment systems event tree are success or failure of the containment overpressure protection function and success or failure of the post accident radioactivity removal function. These outcomes and the early/late core melt state determine the release category assignment for each accident sequence.

After each of the 12 dominant accident sequences were run with each of the 6 containment systems sequences, a further screening was performed resulting in only 18 sequences as candidates for the next recovery analysis. Another recovery analysis is needed since previous recovery analysis could not be conveniently brought forward nor would it necessarily be applicable since the cut sets change in the new Boolean reduction.

2.1.5 Interface with Accident Phenomenology

Each of the 18 sequences to be assigned to release categories consists of an initiating event, a core melt event sequence, and containment systems event sequence. The results of accident phenomenological codes applied in past PRAs were reviewed for apiplicability to St. Lucie and a table developed to map each containment system/early-late core melt state into a release category. This mapping assigns a containment failure mode and apportions the containment failure probability between those modes (the sum equals 1). The containment failure modes are:

α	In-vessel steam explosion
β	Containment leakage
γ	Hydrogen burn overpressure
δ_e	Ex-vessel steam spike
δ_ℓ	Steam and noncondensible gas overpressure
ϵ	Base mat melt through

The sum of the probabilities for the sequences placed in each release category is then combined with the corresponding results from the special emergencies. Finally, the CRAC consequence code was run for the St. Lucie site providing a population dose factor (out to 50 miles radius) for each of the seven PWR release categories. Thus the total population dose (out to 50 miles radius) equals the sum of the probability of release category i * population dose i for i = 1 to 7. This is used in the value-impact analysis.

2.1.6 Value Analysis

The core melt probability and population dose described in the previous subsections comprise the base case. The modifications proposed for each of the vulnerabilities identified can be grouped along with special emergency modifications to form alternatives. The value of an alternative is the decrease in core melt probability or the decrease in population dose from the base case. Therefore the entire process is repeated for each alternative. There are some shortcuts but essentially it is the same method. In the St. Lucie analysis there was only 1 alternative affecting the internal analysis.

2.1.7 Example

It is instructive to follow one sequence from the initiating event all the way through to the population dose with references to the tables or figures in the analysis where each step occurs. This accident sequence is:

$$T_3Q\bar{L}\bar{D}_1H_1'H_2' - \bar{Y}_1\bar{C}_1F'-\alpha$$

This is a transient initiating event T_3 with offsite power and the power conversion system initially available with success of the auxiliary feedwater system ($\bar{L}$), but failure of the pressurizer safety valves (Q) to reclose after opening although not demanded, which leads to a transient induced LOCA (see Figure B.3 branch 6). The sequence then enters the LOCA event tree where the auxiliary feedwater system is successful ($\bar{L}$). The high pressure injection system is successful ($\bar{D}_1$) but both the high pressure recirculation system (H_1') and the low pressure recirculation system (H_2') fail leading to a late core melt. The primes on the H events indicate that the LPRS heat exchangers are not required according to the success criteria, since the AFWS is successful.

Given core melt, the containment system response is success of the containment air recirculation cooling system ($\bar{Y}_1$), success of the containment spray injection system ($\bar{C}_1$), but failure of the containment spray recirculation system (F'), where the prime again indicates that the LPRS heat exchangers are not required. The containment event tree is given in Figure B.5 where Y_1 is composed of the containment fans.

The core melt probability is

$$p(T_3Q) * p(\bar{L}\bar{D}_1H_1'H_2') = (1.5E-2) * (5.5E-5) = 8.3E-7$$

where the multiplier frequency is found in Table B.1 and the Boolean accident sequence probability is found in Table B.11 sequence number 38 or in Table B.12 which is a summary table. The probability of the sequence including the containment systems is

$$p(T_3Q\bar{L}\bar{D}_1H_1'H_2' - \bar{Y}_1\bar{C}_1F') = 7.5E-7$$

found in Table B.28. This value is approximately the same as the core melt probability due to the close coupling of the systems and the fact that the other containment systems states are negligible, i.e., $p(\bar{Y}_1\bar{C}_1F') \simeq 1$.

This is the end of the internal analysis but it is useful to carry the sequence through to its population dose. The sequence state is a late core melt, containment overpressure success ($\bar{Y}_1$) and a late containment spray failure ($\bar{C}_1F'$). Table 7.2 indicates that the α containment failure mode maps into release category 1 with a probability of 1E-4. Thus the complete sequence probability is

$$p(T_3QH_1'H_2'-F'-\alpha) = p(T_3QH_1'H_2'-F')*p(\alpha)$$

$$= (7.5E-7)*(1E-4) = 7.5E-11$$

with the success events not shown as is customary. The corresponding population dose for the baseline source term case (Table 7.5) is (7.5E-9)*(8.5E+5) = 6.38E-5 person rem.

2.2 Core Melt Probability Results

The St. Lucie model started with 64 accident sequences and through several screening steps and adjustments to the model to account for lack of various emergency operating procedures, reduced to 12 accident sequences that were important. It should be kept in mind that under the conditions of this analysis, the base case assumes that feed and bleed is now available. These results are summarized in Table 2.1. The details of the analysis are found in Appendix B. Additional details relative to bleed and feed may be found in Section 10.1. The vulnerabilities identified are described in Section 2.3.

These 12 sequences can be grouped according to the type of failures leading to core melt.

a) T_1MLE
T_2MLE
T_4MLE
T_5MLE - Transient with initial loss of power conversion system, failure of auxiliary feedwater and failure of bleed and feed
p = 8.3E-6

b) $S_2MD_1D_2$ - Small LOCA and failure of ECC injection
p = 1.9E-6

c) $T_1MQD_1D_2$
$T_2MQD_1D_2$
$T_3QD_1D_2$ - Transient induced LOCA and failure of ECC injection
p = 1.6E-6

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Table 2.1

St. Lucie Core Melt Accident Sequences

Dominant Accident Sequences	Probability Before Recovery	Probability After Recovery
T_1MLE	8.3E-5	7.9E-6
$S_2MD_1D_2$	1.4E-5	1.9E-6
$T_3QD_1D_2$	1.1E-5	1.2E-6
$S_2MH_1'H_2'$	1.9E-5	1.1E-6
$T_3QH_1'H_2'$	1.5E-5	8.3E-7
T_1MQLD_1	2.0E-6	2.9E-7
$T_1MQD_1D_2$	1.2E-6	1.9E-7
T_2MLE	9.1E-6	1.9E-7
$T_2MQD_1D_2$	1.5E-6	1.7E-7
T_4MLE	1.2E-5	1.7E-7
$T_2MQH_1'H_2'$	2.0E-6	1.2E-7
T_5MLE	6.4E-6	4.1E-8
	1.8E-4	1.41E-5

d) $S_2MH'_1H'_2$ - Small LOCA and failure of ECC recirculation
p = 1.1E-6

e) $T_2MQH'_1H'_2$,

$T_3QH'_1H'_2$ - Transient induced LOCA with failure of ECC recirculation
p = 9.5E-7

f) T_1MQLD_1 - Transient induced LOCA and failure of auxiliary feedwater and ECC high pressure injection
p = 2.9E-7

The most important sequences for St. Lucie are transients with loss of both auxiliary feedwater and bleed and feed, followed by small LOCAs with a failure of ECC injection, transient induced LOCAs with failure of ECC injection, and small LOCAs with failure of ECC recirculation.

2.3 Potential Internal Vulnerabilities

Ten potential vulnerabilities were identified from the internal analysis for further consideration. Briefly, the procedure was to first examine the cut sets of the dominant accident sequences generated from the fault tree models and generic failure data. A recovery analysis was then performed on all the significant cut sets to give credit for normal operator actions and emergency procedures. The significant cut sets remaining constitute the potential vulnerabilities described here for further examination and possible suggested modifications. This procedure provides a screening process to focus attention on the most significant cut sets and specific basic events in those cut sets.

The following descriptions first identify the cut set which may be one or more basic events and the accident sequences to which that cut set contributes. Each accident sequence is made up of events that were not modeled, called the multiplier, and a Boolean product of events that were modeled. The Boolean expression is made up of numerous cut sets. The probability of each of these cut sets times the probability of the multiplier is a contribution to the internal analysis core melt probability. The cut sets to be discussed below are those cut sets which contribute significantly to the overall core melt probability in one or more accident sequences .

A cut set may appear in more than one accident sequence. A cut set probability is usually the same each time it appears (the exception being where it is dependent on an event in the multiplier expression) but its contribution to the overall core melt probability is the cut set probability times the particular accident sequence multiplier probability. This contribution is given in the vulnerability descriptions to the right of each dominant accident sequence in which that cut set appears. The sum of the contributions from each dominant accident sequence in which that cut set appears represents the total effect of that cut set on core melt probability for the plant based upon the internal analysis. A brief description of the accident sequence events is given in Table 2.2 as a reference.

The description explains the basic events in the cut set and why it is important to core melt. The suggested modification is the instruction to the architectural engineer for an impact analysis. This may have been changed during subsequent discussions regarding feasibility of the modification and validity of the problem identified. In general, the evolution of the modification during the impact analysis is described here. An additional discussion is provided in some cases with any further insights or information that support the reasoning behind a suggested modification.

Table 2.2

Accident Sequence Events

Initiating Events

S_2	Small LOCA < 2 inch Diameter
T_1	Loss of Offsite Power Transient
T_2	Loss of Power Conversion System
T_3	Transients with Main Feedwater Initially Available
T_4	Transients Resulting from Loss of an AC Bus
T_5	Transients Resulting from Loss of a DC bus

System Events

M	Failure of Main Feedwater
Q	Failure of SRVs or PORVs to Reclose Given That They Were Opened
L	Failure of the Aux Feedwater System
D_1	Failure of HPIS
D_2	Failure of LPIS
H_1'	Failure of HPRS Without RHR Heat Exchanger
H_2'	Failure of LPRS Without RHR Heat Exchangers
H_1	Failure of HPRS With RHR Heat Exchangers

INTERNAL VULNERABILITY 1: Common mode failure of the batteries

APPLICABLE CUT SET: BATTERY-CM

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

T_1MLE	2.8E-6
T_1MQLD_1	1.3E-7
	2.9E-6

DESCRIPTION: Following a loss of offsite power initiating event, the batteries are required to start the emergency diesel generators. If the batteries fail to supply the DC power for field flashing, the plant will suffer a loss of all onsite AC electric power. AC power is needed in order to power the high pressure injection pumps for coolant injection (D_1) following a transient induced LOCA as well as for bleed and feed (E). In addition, the pressurizer block valves require AC power to open for bleed and feed if they are normally closed. The two motor operated auxiliary feedwater pumps also run off AC power, while the turbine driven pump has DC powered steam supply and feedwater control valves which would fail closed on a battery common mode event and would have to be manually operated. Therefore, a common mode failure of the batteries would fail the emergency diesel generators and the entire auxiliary feedwater system with possible manual recovery of the turbine driven AFW pump train.

MODIFICATION: No modification is proposed at this time, however, it is recommended that procedures be in place for use of the non-safety related batteries to power safety loads or that the diesel generators be capable of hand-crank starting.

INTERNAL VULNERABILITY 2: Common mode failure of the component cooling water pumps.

APPLICABLE CUT SET: CCW-PUMP-CM

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

$S_2MD_1D_2$	8.0E-7
$T_3QD_1D_2$	6.0E-7
	1.4E-6

DESCRIPTION: Failure of the three component cooling water pumps from a common cause would result in overheating of the HPI pumps and LPI pumps and emergency core coolant injection failure after a small LOCA or transient induced LOCA. In addition, the containment spray injection pumps, fan coolers, and shutdown heat exchangers require component cooling water for successful operation.

MODIFICATION: No modification is proposed at this time.

INTERNAL VULNERABILITY 3: Local fault failures of the containment sump recirculation motor operated valves

APPLICABLE CUT SET: SIS072B-VCC-LF * SIS072A-VCC-LF

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

$S_2MH_1'H_2'$	4.9E-7
$T_3QH_1'H_2'$	3.6E-7
	8.5E-7

DESCRIPTION: Following successful ECC injection in response to a small LOCA or transient induced LOCA, the ECC systems would receive a recirculation actuation signal (RAS) to close the refueling water storage tank valves and open the sump recirculation valves. If these two parallel motor operated valves each fail due to separate local faults, the high pressure, low pressure, and containment spray recirculation systems would fail to receive water. A portion of these valve failures could be locally recovered by manually opening the valves, however, this recovery may only be valid for control circuit or loss of power events and not for mechanical binding.

MODIFICATION: No modification proposed at this time.

INTERNAL VULNERABILITY 4: Common mode failure of the diesel generators combined with local failure of the turbine driven AFW pump

APPLICABLE CUT SET: Diesel-CM * AFW1C-PTS-LF

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

T_1MLE	7.8E-7
T_1MQLD_1	3.7E-8
	8.2E-7

DESCRIPTION: The common mode failure of the diesel generators following a loss of offsite power would result in the loss of all onsite AC electric power. This loss of AC power would fail all ECC injection as well as the motor driven auxiliary feedwater pumps. Subsequent failure of the turbine driven AFW pump due to a local fault would lead to a core melt scenario.

MODIFICATION: Since this event contributes less than 1E-6 to the total core melt probability, no modifications are proposed at this time. However, timely AC power supply from Unit 2 would seem to be a beneficial plant feature. The current time required to make this cross connection is much longer than the calculated time for the Unit 1 steam generators to boil dry.

INTERNAL VULNERABILITY 5: Local failures of the two diesel generators combined with local failure of the turbine driven AFW pump

APPLICABLE CUT SET: DGN1-GEN-LF*DGN2-GEN-LF*AFW1C-PTS-LF

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

T_1MLE	7.6E-7
T_1MQLD_1	3.5E-8
	8.0E-7

DESCRIPTION: Independent local faults of the diesel generators following a loss of offsite power would result in the loss of all onsite AC electric power. This loss of AC power would fail all ECC injection as well as the motor driven auxiliary feedwater pumps. Subsequent failure of the turbine driven AFW pump due to a local fault would lead to a core melt scenario.

MODIFICATION: See internal vulnerability #4.

INTERNAL VULNERABILITY 6: Local fault of diesel generator 1A combined with local failure of DC battery 1B

APPLICABLE CUT SET: DGN1-GEN-LF*BAT125-B-LF

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

T_1MLE	4.3E-7
T_1MQLD_1	2.1E-8
	4.5E-7

DESCRIPTION: Following a loss of offsite power, failure of diesel generator 1A would fail the ECC A train pumps as well as the intake cooling water, component cooling water and AFW train A pumps. Local failure of 125V battery 1B would fail the starting of diesel generator 1B which would, in turn, fail all of the B train pumps as well as the swing pumps. Loss of battery 1B would also fail the DC powered valves in the turbine driven AFW train and would require local recovery of the valves. The loss of offsite power would also prevent recovery of the main power conversion system.

MODIFICATION: Since this event contributes less than 1E-6, no modification is proposed at this time. However, it is strongly

0033

recommended the AFW motor operated valve MV-08-3 remain locked open, and that AFW valves MV-08-14 and MV-09-12 receive their DC power from 125V DC bus 1A. This would add diversity to the turbine driven AFW train and reduce the need for manual recovery of valves.

INTERNAL VULNERABILITY 7: Common mode failure of the intake cooling water pumps

APPLICABLE CUT SETS: ICW-PUMP-CM

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

$S_2MD_1D_2$	2.0E-7
$T_3QD_1D_2$	1.5E-7
	3.5E-7

DESCRIPTION: Common mode failure of the intake cooling water pumps will prevent adequate heat removal via the component cooling water heat exchangers. This, in turn, will lead to overheating of the high and low pressure injection pumps and emergency core coolant injection failure. This failure of the ECC will not occur immediately, however, since the component cooling water system can absorb some of the initial component heat.

MODIFICATION: No modification is proposed at this time.

INTERNAL VULNERABILITY 8: Local fault of diesel generator 1A with test or maintenance unavailability of 125V battery 1B

APPLICABLE CUT SET: DGN1-GEN-LF*BAT125-B-ZBT-UTM

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

T_1MLE	2.6E-7
T_1MQLD_1	1.3E-8
	2.7E-7

DESCRIPTION: Following a loss of offsite power, failure of diesel generator 1A would fail the ECC A train pumps as well as the intake cooling water, component cooling water and AFW train A pumps. A test or maintenance outage of 125V battery train 1B would fail the starting of diesel generator 1B which would, in turn, fail all of the 1B train pumps as well as the swing pumps. Loss of battery 1B would also fail the DC powered valves in the turbine driven AFW train and would require local recovery of the valves. The loss of offsite power would also prevent recovery of the main power conversion system.

MODIFICATION: See internal vulnerability #6

INTERNAL VULNERABILITY 9: Common mode failure of the safety injection system logic due to test or maintenance errors.

APPLICABLE CUT SET: SISLOG-UTM-CM

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

$S_2MD_1D_2$	1.2E-7
$T_3QD_1D_2$	3.0E-8
	1.5E-7

DESCRIPTION: A common mode error made during the testing or maintenance of the safety injection signals could fail the successful automatic initiation of ECC following a small LOCA or transient induced LOCA. The high probability of manual initiation of safety injection from the control room reduces the core melt contribution of this event to 1.5E-7/Rx-yr

MODIFICATION: No modification is proposed at this time.

INTERNAL VULNERABILITY 10: Common mode failure of the recirculation actuation signal logic due to test or maintenance errors

APPLICABLE CUT SET: RASLOG-UTM-CM

Dominant accident sequences affected and estimated core melt frequency contribution resulting from this vulnerability/cut set

$S_2MH_1'H_2'$	4.0E-8
$T_3QH_1'H_2'$	3.0E-8
	7.0E-8

DESCRIPTION: After successful ECC injection, in response to a small LOCA or transient induced LOCA, a low RWST indication will send an automatic recirculation actuation signal (RAS). This signal will realign the valving to allow water to be drawn from the sump for high pressure, low pressure, and containment spray recirculation. If a common mode error is made during the testing or maintenance of the RAS, it could fail the automatic initiation of recirculation. The high probability of manual initiation reduces the core melt contribution of this event to 7.0E-8/Rx-yr.

MODIFICATION: No modification is proposed at this time.

2.4 Summary of Internal Analysis Recommended Modifications

There were ten internal vulnerabilities identified in this analysis but no one event contributed more than 3E-6 to the overall core melt probability. Therefore, no modifications are proposed based upon the insignificant reductions in core melt probability to be gained.

However, engineering judgment suggests that the following recommended modifications be considered:

a. stagger the testing of auxiliary feedwater pumps to reduce the possibility of a common mode error.

b. lock AFW motor operated valve MV-08-03 open permanently.

c. modify AFW valves MV-08-14 and MV-09-12 to receive DC power from 125V bus 1A.

d. provide a diverse method for starting the diesel generators besides the single batteries currently aligned, or

e. provide rapid means for AC electric power from Unit 2.

0036

3.0 SPECIAL EMERGENCY ANALYSES - METHODS AND RESULTS

Task Action Plan A-45 has been developed to assess the adequacy of decay heat removal in existing light water reactors in response to both "internal" and "external" challenges. While internal challenges are generally defined as those probabilistic random failures normally analyzed in a probabilistic risk assessment (PRA), few PRAs have considered the potential contribution to core melt probability due to the "external" events. Since many of these "external" challenges actually occur inside a plant (e.g., fire, turbine-generated missiles, pipe whip), we call them all special emergencies to avoid confusion.

The list of special emergencies that can be postulated to occur is quite extensive and a thorough analysis of each one would require resources beyond the scope of this program. However, reasonable rationale can be applied to exclude many special emergencies from consideration in this program. It should also be noted that combinations of special emergencies from different causes occurring simultaneously, or external events shortly after a random initiating event, are not being analyzed here.

The PRA Procedures Guide (NUREG/CR-2300)[7] lists four bases for excluding special emergencies from detailed analyses. These reasons are listed below:

1) The event is of equal or lesser damage potential than the events for which the plant has been designed. This requires an evaluation of plant design bases in order to estimate resistance to a particular external event. For example, it is established that safety-related structures designed for earthquake and tornado loadings can safely withstand a 1-psi peak positive incident overpressure from explosions. Hence, if the PRA analyst demonstrates that the overpressure resulting from explosions at a source (e.g., railroad, highway, or industrial facility) cannot exceed 1 psi, these postulated explosions need not be considered. It is assumed that the conditional frequencies of failure of structures and components for overpressure of less than 1 psi are negligible given that the safety-related structures are designed for earthquake and tornado loadings. This screening criterion is not applicable to events like earthquakes, floods, and extreme winds since their hazard intensities could conceivably exceed the plant design bases.

2) The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties and could not result in worse consequences than those events. For example, the PRA analyst may exclude an event whose mean frequency of occurrence is less than some small fraction of those for other events; the uncertainty in the frequency estimate for the excluded event is judged by the PRA analyst as

0037

not significantly influencing the total risk. Alternatively, the analyst may decide to compare event occurrence frequencies at some high confidence level (e.g., 95 percent). After the total plant risk is estimated, the deleted external events may have to be reviewed to ascertain that a detailed assessment would not reveal them as significant contributors to the total plant risk.

3) The event cannot occur close enough to the plant to affect it. This is also a function of the magnitude of the event. Examples of such events are landslides, volcanic eruptions, earthquake-fault ruptures (seismic motion and its effects are treated under seismic events), and explosions.

4) The event is included in the definition of another event. For example, storm surges and seiches can be included in external flooding; the release of toxic gases from sources external to the plant is included in the effects of either pipeline accidents, industrial or military facility accidents, or transportation accidents.

For the purposes of TAP A-45, two additional arguments have been developed to exclude certain special emergencies from further analysis.

5) The event does not directly effect the decay heat removal systems and, therefore, the proposed protection does not involve a modification to the decay heat removal system.

6) The event is slow in developing and there is sufficient time to go to cold shutdown, to eliminate the source of the threat, or to prepare an acceptable alternative for the removal of decay heat.

An event deserving further discussion is aircraft impact. Due to the solid construction of plant structures, only a large aircraft would be capable of damaging decay heat removal systems. Aircraft impact is also a low probability event. In addition, in the newer plants the separation (of redundant trains) used to satisfy fire, flood and sabotage concerns also serves to reduce the vulnerability to common mode failure due an aircraft crash. In some older plants, redundant trains may not be as well separated, however, diverse systems (e.g. auxiliary feedwater and high pressure injection) usually are separated so that it is unlikely that a single event would disable all decay heat removal capability. Certainly, the addition of a dedicated decay heat removal system (in separate buildings) to reduce the core melt contribution from those special emergencies which are found to be important contributors to risk, would also reduce the aircraft vulnerability. The impact from aircraft crash debris is similar to tornado missiles, the resulting fire from the crash would be covered by the fire event, and any explosion would be similar to the explosion events. Therefore, aircraft accident analyses will not be pursued further.

Applying the foregoing rationale to the other postulated special emergencies gives the following results:

Event	Applicable Rationale	Remarks
Avalanche	3	Can be excluded for most sites in the United States
Biological Phenomena	6	Sufficient time to prevent damage (assuming a plant awareness)
Coastal Erosion	4	Included in the effects of external flooding
Dam Failure	4	Included in the effects of external flooding
Drought	1,6	Excluded under the assumption that there are multiple sources of ultimate heat sink or that the ultimate heat sink is not affected by drought (e.g., cooling tower with adequately sized basin)
Explosions	1	Tornado missiles, tornadoes, and high winds govern
External Flooding	-	Requires detailed study
Extreme Winds and Tornadoes	-	Requires detailed study
Fire (internal)	-	Requires detailed study
Fog	1	Could, however, increase the frequency of man-made hazard involving surface vehicles or aircraft
Forest Fire	1	Fire cannot propagate to the site because the site is cleared
Frost/Freezing	1	Snow and ice govern
Hail	1	Other missiles govern

Event	Applicable Rationale	Remarks
High Tide, High Lake Level, or High River Stage	4,6	Included under external flooding
High Summer Temperature	1	Ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms
Hurricane	-	Requires detailed study, partially included under external flooding. Wind forces can be covered under extreme winds and tornadoes
Ice Cover	1,4	Ice blockage of river is partially included in flood
Internal Missiles	4	Included under turbine-generated missile
Industrial or Military Facility Accident	5	Included under explosions or toxic gas
Internal Flooding	-	Requires detailed study
Landslide	3	Can be excluded for most sites in the United States
Lightning		Requires detailed study
Low Lake or River Water Level	1	Ultimate heat sink is designed for at least 30 days of operation, taking into account evaporation, drift, seepage, and other water-loss mechanisms

Event	Applicable Rationale	Remarks
Low Winter Temperature	1,6	Thermal stresses and embrittlement are insignificant or covered by design codes and standards for plant design; generally, there is adequate warning of icing on the ultimate heat sink so that remedial action can be taken
Meteorite	2	All sites have approximately the same frequency of occurrence
Pipeline Accident (gas, etc.)	4,5	Included under explosions and toxic gas
Pipe Whip	-	Requires detailed study
Intense Precipitation	4	Included under external and internal flooding
Release of Chemicals in Onsite Storage	4,5	Included under explosions and toxic gas
River Diversion	1,4,6	Considered in the evaluation of the ultimate heat sink; should diversion become a hazard, adequate storage is provided
Sabotage	-	Requires detailed study
Sandstorm	1,5	Included under tornadoes and winds
Seiche	4	Included under external flooding
Seismic Activity	-	Requires detailed study
Ship Collision	2	A loss of the crib house would not immediately threaten the core
Snow	1,4, 5,6	Snow melt causing river flooding is included under external flooding

0041

Event	Applicable Rationale	Remarks
Soil Shrink-Swell tion Consolidation	1	Site-suitability evaluation and site development for the plant are designed to preclude the effects of this hazard
Storm Surge	4	Included under external flooding
Transportation Accidents	4,5	Included under explosions and toxic gas
Tsunami	4	Included under external flooding and seismic events
Toxic Gas	5	Does not threaten DHR systems
Turbine-Generated Missile	-	Requires detailed study
Volcanic Activity	3	Can be excluded for most sites in the United States
Waves	4	Included under external flooding

Therefore, this rather formidable list of special emergencies has been reduced on inspection to the following:

External Flooding
Extreme Winds and Tornadoes
Fire (internal)
Hurricane
Internal Flooding
Lightning
Pipe Whip
Sabotage
Seismic Activity
Turbine-Generated Missiles

The methods for analysis of pipe whip and turbine-generated missiles are beyond the resources of this program. These subjects are considered in the Safety Evaluation Reports (SERs) for individual plants and were considered again in the Standard Review Plan evaluations. Therefore, they will not be considered in the A-45 study. This leaves eight potential special emergencies for further consideration. Each of these special emergencies is addressed separately in this section. A brief

0042

explanation of the analysis methods, quantification of decay heat removal vulnerabilities to these events, and a description of potential modifications are given. The complete analysis details are provided in the referenced appendix.

It should be realized that the quantification of the special emergencies has large uncertainties in many cases. Since the state of the art is not very advanced in the analysis of many special emergencies, it was necessary to make assumptions in order to perform the calculations. These assumptions are documented in the following analysis summaries as well as in the appropriate appendices. Nonetheless, a residual risk may remain for such special emergencies as fire, flood, and earthquake.

3.1 Seismic Analysis

Seismic events have a wide range of factors which affect their threat to nuclear power plants. The location of the earthquake, the magnitude of the ground acceleration, the soil transfer of energy, the elevation of plant equipment, and the seismic equipment supports are only a few of the factors to be considered in an analysis of plant vulnerabilities. However, the fact that earthquakes in the United States have affected large portions of the country requires an analysis of the probability and uncertainties of a resulting core melt.

Successful completion of this program required a simplified seismic risk assessment methodology which could be applied to particular plants being studied in a far shorter time than previous PRA studies. This section describes the procedures developed for the simplified seismic analysis and presents the results of the risk quantification for the St. Lucie Nuclear Power Plant.

A more detailed description of the seismic analysis is presented in Appendix C of this report.

3.1.1 Analysis Methods

There are seven steps required for calculating the seismic risk at a nuclear power plant:

1) Determine the local earthquake hazard (Hazard Curve and Site Spectra).

2) Identify accident scenarios for the plant which lead to radioactive release (Initiating Events and Event Trees).

3) Determine failure modes for the plant safety and support systems (Fault Trees).

4) Determine fragilities (probabilistic failure criteria) for the important structures and components.

0043

5) Determine the responses (accelerations or forces) of all structures and components (for each earthquake level).

6) Compute the probability of core melt using the information from Steps 1 through 5.

7) Estimate uncertainty in the core melt probabilities.

Only the level of detail differentiates a simplified seismic analysis from a detailed seismic PRA. The seven steps of the TAP A-45 simplified seismic risk analysis procedure are described below.

Step 1 - Seismic Hazard Characterization -

a) If a seismic hazard curve exists for the plant (e.g., from the Systematic Evaluation Program (SEP) program or from an existing PRA of a nearby plant) it is used.

b) If no such curve exists, a hazard curve scaled to an exceedance probability of 2.5E-4/yr at the safe shutdown earthquake (SSE) will be used. The slope of the curve for higher peak ground acceleration (PGA) values is estimated from other hazard curves for the same broad seismological province.

c) All plants west of the Rocky Mountains require site specific hazard curves due to the high levels of seismic activity, but most existing plants have hazard curves already available.

Step 2 - Initiating Events and Event Trees - The scope of TAP A-45 includes only the initiating events associated with small loss of coolant accident (SLOCA) and transient events. Two types of transients are being considered: those in which the power conversion system (PCS) is initially available (denoted Type T_3 transients) and those in which the PCS is failed as a direct consequence of the initiating event (denoted Type T_2 transients). The event trees derived for the internal event analyses for TAP A-45 are utilized.

The small LOCA initiating event frequencies are being computed based on the statistical distribution of small pipe failures computed as part of the NRC-sponsored Seismic Safety Margins Research Program (SSMRP).[2] The restriction to small LOCAs is not particularly limiting, because existing PRAs have shown that large and medium LOCAs are not major contributors to total risk of core damage.

The frequency of Type T_2 transients is based on the probability of loss of offsite power (LOSP). This will always be the dominant cause of these transients (for the majority of plants for which LOSP causes loss of main feedwater). The Type T_3 initiating event is computed from the condition that the sum of the initiating event probabilities considered must be unity.

The hypothesis is that, given an earthquake of reasonable size (3.75 magnitude Richter was used in the SSMRP), at least one of the initiating events will occur.

__Step 3 - Fault Trees__ - Simplified systems fault trees for a number of example plants have been developed as part of TAP A-45, (for random failures only). These fault trees will be used, although they require some modification to include basic events for seismic failure modes and re-solving the trees for pertinent cut sets to be included in the probabilistic calculations. Probabilistic culling is used in re-solving these trees to assure that important correlated failure modes are not lost.

__Step 4 - Component and Structure Failure Descriptions__ - Component seismic fragilities are obtained both from a generic fragility data base and from plant specific fragilities derived for components identified during a plant walkdown.

The generic data base of fragility functions for seismically-induced failures was originally developed as part of the SSMRP[3]. Fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data utilized in developing fragility curves were obtained from the results of component manufacturer's qualification tests, independent testing lab failure data and data obtained from the extensive U.S. Corps of Engineers SAFEGUARD Sub-system Hardness Assurance Program. These data were then statistically combined with the expert opinion survey data to produce fragility curves for the generic component categories. Inasmuch as four years have passed since this data base was originally developed, the site-specific fragilities developed for some 20 PRAs performed in the interim were examined, and compared[4] against the original data base. This resulted in modifications of median fragility levels of 5 of the original 37 generic categories. For tipping and sliding of cabinets, and for dynamic analyses of tanks, graphical methods of analysis have been developed to greatly simplify the required fragility development.

Within the time scale prescribed, structural fragility development is not possible, and could not be included for TAP A-45. This is appropriate since it is the design requirements for decay heat removal systems which are being evaluated, and not the structural design. However, the approach being described here should prove widely useful in examining other generic and value/impact issues not directly concerned with adequacy of power plant structures.

__Step 5 - Seismic Response of Structures and Components__ - Building and component seismic responses are estimated from peak ground accelerations at several probability intervals on the hazard curve. Three basic aspects of seismic response--best estimates, variability, and correlation--are estimated. SSMRP Zion analysis results and simplified methods studies form the basis for assigning scaling, variability and correlation of responses.

In each case, SHAKE code[2] calculations are performed to assess the effect of the local soil column (if any) on the surface peak ground acceleration and soil-structure interactions. This permits an appropriate evaluation of the effects of non-homogeneous underlying soil conditions which can strongly affect the building responses.

Two situations are commonly encountered in developing structural responses. First, for many early plants, only design floor spectra are available from the design reports. For this situation, the results of a detailed soil-structure interaction investigation[5] made as part of the SSMRP may be used to scale the computed design floor slab accelerations to best estimate values including the effects of the soil; and to estimate best estimate floor response spectra.

Second, one has the situation where fixed-base mass-spring or eigen-system descriptions are available for one or more buildings at the site. For this case, one can compute the floor slab accelerations using the CLASSI code[2]. This code takes a fixed base eigen-system model of the structure and input-specified frequency dependent or independent soil impedances and computes the structural response (as well as variation in structural response if desired).

Variability in responses (floor and spectral accelerations) is assigned based on the SSMRP results. The recommended uncertainties (expressed as standard deviations of the logarithms of the responses, β) are shown below:

Quantity	β random
Peak Ground Acceleration	0.25
Floor Zero Period Acceleration	0.35
Floor Spectral Acceleration	0.45

Correlation between component failures is being included explicitly in TAP A-45. In computing the correlation between component failures (in order to quantify the cut-sets) it is necessary to consider correlations both in the responses and in the fragilities of each component. Inasmuch as there are no data as yet which show correlation between fragilities, the fragility correlations between like components are taken as zero and 1, and the possible effect of such correlation quantified in a sensitivity study. The correlation between responses is assigned according to a set of rules given in Reference 1. These are applied to both BWR and PWR plants.

Step 6 - Probabilistic Failure and Core Melt Calculations - Given the input from the five steps above, the SEISIM[2] and SETS[1] codes are used to calculate the required output (probabilities of failure, core melt, etc.). Use of the SEISIM code for the final computation of accident sequences permits proper inclusion of seismically-induced correlation between component failures.

Step 7 - Estimate Uncertainties - Uncertainties in core damage estimates are being based entirely on upper and lower confidence limits for the hazard curve. Generic recommendations for such limits are given in Reference 1. This simplification is appropriate since previous seismic PRAs have shown the hazard curve to contribute 70-90% of the total seismic risk uncertainty.

3.1.2 Seismic Vulnerabilities

The safe shutdown earthquake for St. Lucie has an acceleration of 0.1 g, however TAP A-45 is examining the vulnerability of decay heat removal systems to initiating events which are beyond the design basis. Therefore, probabilities of seismically induced core melt were calculated for earthquakes in the ranges of 1-2 SSE, 2-3 SSE, 3-4 SSE, and above 4 SSE. For the St. Lucie site, the frequency of exceedance for earthquakes of these magnitudes are estimated in Appendix C as follows:

PGA Level	Exceedance Frequency/yr
0.1	2.5E-4
0.2	5.0E-5
0.3	1.9E-5
0.4	8.0E-6

An earthquake may initiate a core melt scenario by causing one of the following plant states: a small loss of coolant accident (S_2), a transient in which the power conversion is initially available (T_3), and those transients in which the power conversion system has failed as a direct consequence of the initiating event (T_2). The frequency of Type T_2 transients is based on the probability of loss of offsite power since this will always be the dominant cause of these transients. While the probability of a small LOCA or a loss of offsite power increase as the earthquake level increases, the probability of a Type T_3 transient will decrease accordingly.

Since all of the plant systems feel the effects of an earthquake, the quantification of core melt scenarios involves calculating component fragilities for the various earthquake levels and using these values in the fault trees developed in the internal event analyses. Table 3.1 summarizes the probability of each of the seismically induced core melt sequences for the varying earthquake levels. The total core melt probability is

0047

Table 3.1 Sequence Core Melt Probabilities for Varying Earthquake Magnitudes - Base Case

	Earthquake Magnitude (Gravities)				
Sequence	0.100 - 0.200	0.200 - 0.300	0.300 - 0.400	Above 0.400	Total
$S_2\bar{L}\bar{D}_1H'_1H'_2$	0.00E+00	0.00E+00	0.00E+00	3.60E-12	3.60E-12
$S_2\bar{L}D_1\bar{X}\bar{D}_2H'_2$	0.00E+00	0.00E+00	0.00E+00	9.60E-15	9.60E-15
$S_2\bar{L}D_1\bar{X}D_2$	0.00E+00	5.59E-09	2.69E-08	2.16E-11	3.25E-08
$S_2\bar{L}D_1X$	0.00E+00	3.65E-09	3.16E-09	9.60E-12	6.82E-09
$S_2L\bar{D}_1\bar{P}_1H_1$	0.00E+00	0.00E+00	8.09E-09	1.12E-13	8.09E-09
$S_2L\bar{D}_1P_1$	0.00E+00	1.69E-08	4.09E-08	3.20E-11	5.78E-08
S_2LD_1	1.75E-09	1.02E-07	3.14E-07	1.36E-10	4.18E-07
$T_2L\bar{P}\bar{Q}EH_1$	0.00E+00	2.63E-09	4.54E-09	6.80E-11	7.24E-09
$T_2L\bar{P}QE$	6.36E-07	4.59E-06	4.07E-06	6.40E-08	9.36E-06
T_2LP	0.00E+00	0.00E+00	0.00E+00	2.16E-13	2.16E-13
$T_3ML\bar{P}\bar{Q}\bar{E}H_1$	0.00E+00	1.54E-09	0.00E+00	4.00E-11	1.58E-09
$T_3ML\bar{P}\bar{Q}E$	4.55E-07	2.20E-06	7.19E-07	4.40E-08	3.42E-06
T_3MLP	0.00E+00	0.00E+00	0.00E+00	1.60E-13	1.60E-13
TOTALS	1.09E-06	6.93E-06	5.19E-06	1.08E-07	1.33E-05

calculated to be approximately 1.33E-5/R-yr.

It can be seen that the major contribution to core melt comes in the two to four SSE range. This is because St. Lucie has an SSE of only 0.1, which is roughly half of the design basis acceleration of most plants but the plant is also located in a low seismic hazard region.

3.1.3 Potential Modifications

The refueling water storage tanks as well as the condensate storage tanks all have a large height-to-diameter ratio in terms of seismic fragility. The refueling water storage tank provides water for the high pressure, low pressure, and containment spray injection pumps. The condensate storage tanks are crosstied and feed the two motor-driven auxiliary feedwater pumps as well as the turbine-driven pump dedicated to Unit 1. In order to prevent core melt following an earthquake which buckles both the RWST and CSTs, the seismic modification involves welding 4 inch wide vertical flanges to the bottom 10 feet of the refueling water storage tank at approximately eight places around each tank. An additional horizontal brace would then be welded to the vertical flanges about 5 feet above the ground to prevent buckling. The CST would be strengthened with spaced braces to the surrounding reinforced missile shield wall.

Table 3.2 presents the recalculated core melt probabilities for the varying earthquake levels and core melt sequences if these modifications were to be implemented. The total seismically induced core melt probability with these modifica- tions incorporated would be approximately 9.61E-7/R-yr.

3.2 Fire Analysis

Based on plant operating experience over the last 20 years, it has been observed that nuclear power plants will probably have three to four significant fires over their operating lifetimes (i.e., on the order of one fire every ten years for a frequency of .1 fire/R yr). Previous probabilistic risk assessments have shown that fires are a significant contributor to the overall core melt probability, contributing from 7% to 50% of the total core melt probability.

Large strides have been made in the fire protection for nuclear power plants in recent years. In particular, 10 CFR 50, Appendix R and Branch Technical Position 9.5-1 have led to improved fire protection measures at nuclear power plants. However, even with these improvements, there remains a residual risk from fires which has not yet been quantified with the exception of those plants that have performed fire-related PRAs. The purpose of this section is to quantify the risks due to fires at the St. Lucie Nuclear Power Plant and to evaluate the value of potential modifications.

Table 3.2 Sequence Core Melt Probabilities for Varying Earthquake Magnitudes - with Modifications

	Earthquake Magnitude (Gravities)				
Sequence	0.100 - 0.200	0.200 - 0.300	0.300 - 0.400	Above 0.400	Total
$S_2\bar{L}\bar{D}_1H'_1H'_2$	0.00E+00	2.26E-09	1.53E-08	6.80E-12	1.76E-08
$S_2\bar{L}D_1\bar{X}\bar{D}_2H'_2$	0.00E+00	0.00E+00	0.00E+00	1.60E-14	1.60E-14
$S_2\bar{L}D_1\bar{X}D_2$	0.00E+00	1.50E-08	1.24E-07	2.16E-11	1.39E-07
$S_2\bar{L}D_1X$	0.00E+00	0.00E+00	6.29E-09	8.00E-13	6.29E-09
$S_2L\bar{D}_1\bar{P}_1H_1$	0.00E+00	0.00E+00	0.00E+00	2.72E-15	2.72E-15
$S_2L\bar{D}_1P_1$	0.00E+00	0.00E+00	3.00E-09	2.96E-13	3.00E-09
S_2LD_1	0.00E+00	2.83E-09	4.65E-08	2.72E-12	4.93E-08
$T_2L\bar{P}\bar{Q}EH_1$	0.00E+00	0.00E+00	0.00E+00	7.20E-13	7.20E-13
$T_2L\bar{P}QE$	1.01E-08	1.09E-07	5.09E-07	9.60E-10	6.29E-07
T_2LP	0.00E+00	0.00E+00	0.00E+00	1.68E-15	1.68E-15
$T_3ML\bar{P}\bar{Q}\bar{E}H_1$	0.00E+00	0.00E+00	0.00E+00	5.92E-13	5.92E-13
$T_3ML\bar{P}\bar{Q}E$	9.24E-09	4.56E-08	6.11E-08	9.60E-10	1.17E-07
T_3MLP	0.00E+00	0.00E+00	0.00E+00	1.44E-15	1.44E-15
TOTALS	1.93E-08	1.75E-07	7.65E-07	1.95E-09	9.61E-07

A more detailed description of the fire analysis is provided in Appendix D of this report.

3.2.1 Analysis Methods

This analysis will evaluate the residual risk from fires that could affect decay heat removal systems. In doing so, existing procedures used in published fire PRAs will be employed, but simplifying assumptions will be made to allow the analysis to proceed without requiring the level of detail required for these other fire PRAs.

Most of the simplifying assumptions are based on insights gained from the previous PRAs or from the historical fire data base. These assumptions are as follows:

a) Only fires in a single area (room) will be considered since only single areas have been seen to be significant contributors to core melt probability. Because no room to room issues are being considered, barriers between adjacent areas are assumed to remain intact during a fire.

b) Fire frequency and suppression reliability will be based on available industry data. The fire frequencies used will be the mean value of generic data collected from fire PRAs. Suppression system data is based on American Nuclear Insurers data and on the HTGR fire methodology PRA.

c) Two transient combustible exposure fires are assumed to bound all transient and electrically-initiated fires. A trash can size (30 gallons) container of refuse with an energy output of approximately 500KW for 15 minutes is one exposure fire used in the computer calculations. The other combustible is a 10 gallon spill of acetone spread over a square area with sides of 2 meters. This fire results in an intensity of 4.67MW for approximately three minutes. Electrically-initiated fires are not being explicitly evaluated, but it is assumed that the exposure fires postulated for this analysis bound any transient or electrically-initiated fire that might occur at a plant. The worst case geometries are used in assessing fire growth and spread and the fire code COMPBRN is used to determine if the cables or components of interest would be damaged by the fire.

d) St. Lucie transient event trees will be used to determine the systems components, from a success-oriented outlook, that need to be considered. Transient event trees are used in this analysis since it is assumed that a fire occurs independently of any initiating event and the operators will always insure that at least a transient will occur once they know that there is a fire (i.e. scram the reactor).

e) Components affected by a fire are electrical/ active components in nature (e.g., cables, motors, switchgear, buses).

Only temperature will be used to evaluate a component's vulnerability to a fire. Passive components, such as pipes and manual valves will be assumed to be unaffected by a fire.

f) Random failures of those systems not directly affected by the fire at a particular location will be considered if applicable.

g) Offsite power is assumed to be available during a fire unless the fire is capable of causing a loss-of-offsite power.

h) Fire growth and spread and suppression of fires are treated as independent of each other.

i) No explicit core melt timing is evaluated.

j) Where exact locations of cables cannot be accurately determined, assumptions as to their locations are made.

The basic approach used in determining the risk from a fire uses the following four basic steps:

a) Initial screening for potentially important fire areas.

b) Plant visit for verification.

c) Quantification of risk.

d) Proposed fixes to reduce the fire risk in affected areas.

The initial screening determines which areas of the plant contain enough front line or support systems in their boundaries such that fire damage could invalidate all of the success branches on the transient event trees. These transient event trees identify the success criteria of the frontline systems or combinations of systems that prevent a core melt scenario. The internal systems analysis is also used as a starting point for the fire analysis to identify the success criteria for such support systems as cooling systems and the AC/DC power requirements. Plant schematics and piping diagrams supplement this information where appropriate. Available documentation then allows screening of fire areas to determine the small set of areas where a single fire could lead to a core melt.

The plant is then visited to see first hand what the physical arrangements are in each of the vital areas. The locations of major fuel sources and some estimate of the coordinates of components and cables are obtained.

In the quantification step, the deterministic fire code COMPBRN is used to ascertain if an exposure fire could damage the components/cables of interest in the area. An event tree is utilized to probabilistically estimate the risk of core melt from a fire in those areas not eliminated by the code calculations. The basic elements of the event tree are as follows:

a) Fire occurrence frequency for the area of interest.

b) Suppression system probability (both for automatic and manual suppression).

c) Operator error and random failure of unaffected systems from a remote shutdown location.

Based on the quantification steps, engineering modifications are proposed for those areas which have a fire-induced core melt probability contribution of greater that 1.0E-6. For any proposed modification, a new core melt probability is calculated by modifying the event tree to reflect the fix. This new core melt probability indicates the "value" of a modification in terms of core melt reduction.

3.2.2 Fire Vulnerabilities

Using the transient event trees, the following systems and success criteria were identified as crucial to preventing core damage at St. Lucie:

a) Power Conversion System: 1 out of 2 trains

b) Auxiliary Feedwater System: 1 out of 3 pump trains (2 motor driven and 1 turbine driven)

c) Safety Injection System: 1 out of 3 trains

d) Residual Heat Removal System (in recirculation mode): 1 out of 2 RHR pumps in series with 1 out of 2 RHR heat exchangers

The success criteria for the support fluid systems are:

a) Intake Cooling Water: 1 out of 3 ICW pumps

b) Component Cooling Water: 1 out of 3 CCW pumps with 1 out of 2 CCW heat exchangers

Because, in general, the success criteria only require that 1 train or pump from each system is needed, to invalidate the success criteria implies that locations must be found in the plant where all trains of a system can be damaged by a fire. More importantly, this means, in general, that for this analysis only those locations where all of the trains of the systems listed above can be damaged are important.

Appendix D presents a review of each plant area to determine the potential for a core melt following a fire in this area. This analysis pointed to only the cable spreading room requiring a more detailed review.

Cable Spreading Room - This zone houses the Unit 1 reactor protection cabinets, battery chargers, inverters and distribution panels for DC buses 1A, 1B and 1AB, as well as other test equipment. The only other items found in this zone are the instrumentation and control circuit cabling for both safety-related and nonsafety-related systems in the plant. This includes the intake cooling, component cooling water, safety injection, RHR, auxiliary feedwater, and PCS systems that are of interest. The zone has a three hour fire rating (roof, floor, walls, doors, penetrations). Fire protection systems found in this zone include (a) photoelectric smoke detectors, (b) automatic, single failure proof, Halon suppression system, and (c) portable CO_2 fire extinguishers. In addition, a three-hour rated barrier has been built around the "A" battery charger, inverter, and DC distribution panel. Cables are routed into cable trays above this enclosure. Located outside this zone are manual hose stations. This zone is a large enclosure and as such, because of its function, has a large amount of cabling overhead (cable trays stacked 6 high, several stacks next to each other). Only a limited number of power cables enter this room.

The fire occurrence frequency for this room is taken directly from the generic fire frequency data base to be 6.7E-3/R yr. The COMPBRN fire code determined that an acetone fire could ignite overhead cables in about 240 seconds while damaging redundant cables across the room in about 900 seconds. The automatic Halon suppression system has a failure probability of 0.2, while the failure probability of manual suppression in 15 minutes (900) seconds is 0.3. To overcome the loss of control of systems due to a fire in this cable spreading room, operators would use the Hot Shutdown Panel (HSP). The failure probability of lining up the ASP in the predicted confusion is assumed to be 0.1. Even if the system is properly aligned, the turbine driven pump train which can be operated from the HSP has a failure probability of approximately 1.0E-2/demand.

The core melt probability for this area, then, is:
(6.7E-3)(.2)(.3)[(.1) + (.9)(1.0E-2)] = 4.38E-5/R yr.

Control Room - This zone contains instrumentation and control circuitry for all redundant safe shutdown systems and other operating systems such as the main power producing turbine generator. Also found in this zone are test and calibration circuits. The bulk of the combustibles in this zone are due to either cabling or transient combustibles. Cabling is found inside steel cabinets and in general enters and exits the control room via sealed floor penetrations. This zone has a three hour fire rating. Fire protection systems for this zone include (a) photoelectric smoke detectors, (b) manual fire extinguishers, and (c) a dedicated smoke removal system. Located outside this zone are manual hose stations. In this zone, the safety injection, turbine driven auxiliary feedwater, intake cooling water pumps, and component cooling water pump controls are found on different control board sections.

The generic fire occurrence frequency for the control room is 4.9E-3/R yr.

In this zone, the systems that are of interest have their controls on various control room panels that are separated from each other. The separation between the panels is on the order of 20 linear feet. This means that for the fire to cause damage to the systems listed above would require the fire to spread a minimum of this distance. In addition, because the cabling of interest only comes up about a meter into the room (to the sloped portion of control boards or about waist high in vertical cabinets), hot gas layer effects would not be expected to play a major part in causing damage; only the direct impingement of a flame or radiation would cause damage. Because the safety system cables are not in the direct line of sight (i.e., cables cannot "see" the other safety-related cables due to obstruction from cabinet walls and supports), the time for flame or radiation damage to occur would be, in all likelihood, long. In addition, because this zone is continually manned, the chance of a fire spreading to a size where it could cause this type of damage is deemed to be small. Because not all the systems that could be used to provide decay heat removal are located on the same panel, and consistent with the assumptions in this analysis and the rational just presented, further analysis of this zone is not warranted.

In considering CR fires, only obvious fire vulnerable cabinets were considered (i.e., those containing front line safety systems). Subtle interactions, such as spurious operation of components, effects of ventilation, and damaging effects which suppression may cause, were ignored consistent with the original set of assumptions.

3.2.3 Potential Modifications

At St. Lucie, it appears that only the cable spreading room would need improvements to the fire protection options already employed in order to reduce the risk of core melt due to fire.

The proposed modification consists of wrapping all train B cabling in a fire wrap to ensure that the train B safety systems can be used for some time before having to shift control to the HSP.

This is the simplest solution because it ensures that B train safety systems will be available for some time given a fire in the zone and because application of a thermal wrapping, once cables have been identified, is a fairly straightforward task.

Other solutions were rejected for the following reasons:

Construction of barriers to isolate portions of the area. This would have entailed putting in a full height wall into the area. The number of penetrations that would be required through this wall would have been large and would potentially require rerouting of cables to ensure redundant systems were on opposite sides of the barrier.

Rerouting of affected cables. This option was not a viable solution since a large amount of engineering work would be required to determine the new routing and still meet Appendix R requirements. In addition, the actual rerouting process would be difficult and time consuming to perform and would have to be done while the plant was shut down, thus potentially affecting its capacity factor if the work took longer than expected.

Automation of manual systems. Not applicable for this zone.

Move components out of the area. Only cables were of interest in this zone, therefore this option is similar to the rerouting of cables option which did not appear to be a viable solution.

Procedural changes. A fire in this zone potentially affects a large number of safety systems by causing electrical failure of these systems. Procedural changes would have little impact on mitigating the potential damage.

Install an additional suppression system. For independence, a water system would be required. However, at St. Lucie no automatic water systems have been used at any location in the plant. Installation of such a system would therefore be prohibitively expensive and the cost benefit derived would consequently be low.

So, for this zone, the best modification appears to be wrapping B train cables. The modified core melt probability obtained with this modification is determined as before.

By wrapping the B train cables with a fire retardant material, their survivability is enhanced. The fire retardant wrap to be used is rated as a one-hour fire barrier (the fire wrap might provide protection in excess of an hour, which would further reduce the core melt frequency, however its survival is not documented past an hour, so for conservatism this value is used). Therefore, the plant has essentially an hour to put the fire out before damage occurs. A fire lasting this long in the cable spreading room without being brought under control by either the automatic Halon system or from the plant's fire fighting brigade is probably an incredible event. However, any fire lasting less than an hour, because of the close proximity of control cables in the cable spreading room, would necessitate the use of the HSP almost immediately without a fire wrap around train B cable. It is therefore assumed that the time for manual suppression (as an upper bound) in this room is one hour. The probability that the fire would not be manually suppressed in this time is .1. (It is assumed that the combination of wrapping the B train cables as well as the extra time this provides for manual suppression ensures that the HSP can be used to control the plant.)

As previously discussed, because the fire that occurs in the cable spreading room has serious consequences, and because there would in all likelihood be a highly confused state at the plant after the occurrence of a fire of this type, a human failure probability of .1 is assumed for properly lining up the HSP.

In addition, given that the motor-driven auxiliary feedwater pump is lined up and placed in service properly to facilitate decay heat removal, there is still a chance that a random failure of the system could occur. A nominal random failure probability for just the one train of motor-driven auxiliary feedwater is approximately 1.0E-01/demand.

Modified core melt probability with a fire retardant wrap around the B train cables:	1.46E-05/R yr

Details of this fire analysis are given in Appendix D along with the analysis event tree methodology.

3.3 Internal Flood Analysis

Operating experience at nuclear power plants has shown that the potential for damage to safety-related equipment as a result of internal flooding is of concern. Examples of internal flooding initiators that have occurred include operators overfilling water tanks, in-leakage of ground water, hose and weld ruptures, pump seal leaks, improper maintenance procedure, and various other circumstances. Several of these events resulted in damage to high pressure coolant injection systems, reactor building spray pumps, service water pumps, emergency feedwater pumps, diesel generator control cabinets and other redundant and diverse safety-related components. Although to date, none of these floodings have led to a serious accident, there is still a risk that needs to be evaluated to determine what the chances are that an internal flood, despite a conscious effort to design the plant to withstand this threat, could lead to a core melt scenario. The TAP A-45 program specifically addresses these concerns including the vulnerability of the

0057

decay heat removal systems to equipment damage due to spray from ruptured water piping and the sections that follow will represent the methodology and assumptions employed to carry out these analyses. More details regarding this analysis are provided in Appendix E.

3.3.1 Analysis Methods

The methods used to analyze internal flooding in TAP A-45 follow the existing procedures employed in published PRAs. However, since a full-scope PRA is beyond the resources of this program, a number of simplifying assumptions have been made. Most of these assumptions are based on insights derived from previous PRAs or by consideration of the physical constraints of a flood in a building. The major assumptions are as follows:

a) Internal flood frequencies, component damage thresholds, and physical room considerations will be based on previous industry wide data bases. The flood sizes have been grouped according to the flow rate from a flood source and estimates have been made for the room drainage and equipment volume based on the Seabrook PRA assumptions.

b) St. Lucie transient event trees are used to determine the systems/components, from a success- oriented outlook, that need to be considered. Transient event trees are used in this analysis since it is assumed that an internal flood occurs independently of any initiating event and the operators will always insure that at least a transient will occur once they know that there is a flood (i.e. scram the reactor). LOCA event trees will not be examined since pipe ruptures have already been considered in these trees.

c) Front line mitigation systems (such as high and low pressure injection) will not be considered as flood sources. This simplifying assumption is based on the fact that these front line systems, 1) have been designed to higher standards, 2) have controlled chemistry, 3) in general do not see large temperature gradients, and 4) have been involved in few catastrophic failures.

d) Flood barriers maintain their structural integrity.

e) Flooding via drains backing up into other areas will not be considered.

f) Only single flood sources will be analyzed.

g) Components will fail from submergence in water to a critical point or from direct spray from the flood source. It is assumed that switchgear and buses will fail if they are standing in six inches of water or more. An electric motor will be assumed to fail when the water level reaches the bottom of the motor casing.

h) Random failures of those systems not directly affected by the internal flood at a particular location will be considered if applicable.

i) Offsite power is assumed to be available during an internal flood unless the flood is capable of causing a loss of offsite power.

j) No explicit core melt timing is evaluated.

The approach used in determining the risk due to internal flooding at the St. Lucie Nuclear Power Plant uses the following four basic steps:

a) Initial screening for potentially important internal flood levels.

b) Plant visit for verification.

c) Quantification.

d) Proposed fixes to reduce the risk due to internal flood in affected areas.

The initial screening involves using the transient event trees to identify the front line and support systems needed to prevent core melt and their success criteria. Once the systems of interest are identified, all available documentation is analyzed to determine those areas where enough components could be disabled by an internal flood so as to cause core damage.

A plant visit is then conducted to see the physical arrangement of equipment, verify the accuracy of documentation, and obtain the plant location coordinates of vital components.

The step of quantifying the contribution to core melt begins with examining whether a flood source could damage the components of interest and the time available for operator mitigation of the event. With the flowrate, room volume, and capacity for water removal known, the time to damage of components may be calculated based on the water depth in the room and the assumptions governing component damage thresholds. Spray is considered when a flood initiates in a room and the critical components are in a direct line of sight with the flood source. An event tree is then constructed to quantify the area flood damage scenario following the following steps:

a) Internal flood occurrence frequency for the area.

b) Probability of detection of the flood.

c) Isolation potential of the flood source.

d) Operator error and random failures of unaffected systems.

Based on the quantification steps, engineering fixes are proposed for these areas which have an internal-flood-induced core melt probability contribution of greater than 1.0E-6. For any proposed fix, a modified core melt probability is then calculated (by modifying the event trees to reflect the fix) to indicate how much the core melt probability could be reduced.

3.3.2 Internal Flood Vulnerabilities

The St. Lucie transient event trees were used to identify the decay heat removal system combinations which are needed to prevent core damage. These systems and their success criteria are summarized below:

a) Power Conversion System (PCS): 1 out of 2 trains
b) Auxiliary Feedwater System: 1 out of 2 motor driven trains or 1 out of 1 turbine driven trains
c) Safety Injection System: 1 out of 3 trains
d) Residual Heat Removal System (in recirculation mode): 1 out of 2 RHR pumps in series with 1 out of 2 RHR heat exchangers

The success criteria for the support fluid systems are:

a) Intake Cooling Water: 1 out of 3 ICW pumps
b) Component Cooling Water: 1 out of 2 CCW pumps with 1 out of 3 CCW heat exchangers

For the frontline systems the success criteria has an important implication for this analysis. Because in general the success criteria only require that 1 train or pump from each system is needed, to invalidate the success criteria implies that locations must be found in the plant where all of a particular system can be damaged by a flood. More importantly, this means in general that for this analysis, only those locations where all of the trains of the systems listed above can be damaged are important. However, locations may exist where a front line system may not have all its trains together, but its support systems are all in near proximity and are vulnerable to a common flood.

The St. Lucie analysis did assume credit for a bleed and feed mode of operation. Therefore, credit is assumed for the use of the safety injection or RHR pumps following a transient initiated by a tank, pipe or valve rupture. Appendix E provides a detailed matrix of the vulnerable plant areas and system locations which was used in the screening of potentially important flood areas. The utility's latest Appendix R submittal and fire emergency plans were useful in mapping the components and their cabling through areas of concern.

The matrix presented in Appendix E eliminates several areas from further investigation since they do not contain sufficient components of concern. Much of St. Lucie is open to the outside which prevents water buildup in an area.

St. Lucie's physical layout can best be described as a detached system of structures. Specifically, the core of the plant is the reactor containment building. An auxiliary building, next to the containment, contains the RHR, SI, and charging pumps as well as associated heat exchangers. The auxiliary building has, in essence, a large basement area where all these pumps are located. There are curbs installed at the entrance to each pump room which would impede the flow of water. All pumps are mounted on concrete pedestals at least a foot high so that a considerable volume of water would be required to damage them due to submergence. However, because of the large area involved, a "large" volume of water would be required to cause any damage.

A control structure, housing the control room, cable spreading room, battery rooms and other electrical equipment is located on top of the auxiliary building structure. These areas are above tanks found at St. Lucie. The only water found in these areas is for firefighting purposes (i.e., manual hose systems, no automatic water suppression systems).

The main power producing turbine generators are located in a separate structure. The "main turbine deck" contains the main turbine and its associated systems (i.e., condensate, feed, etc.). This structure has multiple openings to the outside so that any internal flood would not be able to build up any appreciable water level. Also found in this general area are the auxiliary feedwater pumps which are essentially out in the open.

The physical separation between the various safety components of interest for this analysis is on the order of several hundred feet. Specifically, the AFW pumps, and SI and RHR pumps, are widely separated, and because the AFW pumps are housed in structures that are open to the outside, any potential internal flood would not have the capability to physically damage both of these systems. (In a similar fashion, the same can be said for the PCS and AFW systems.) (Note: a flood in the auxiliary building, if located near or in the RHR pump rooms, could damage these pumps since the RHR pump rooms are at the lowest level in the plant. However, the AFWS and PCS success paths would not be affected.)

Support systems for front line systems (i.e., CCW and ICW) are all located in outside areas. The CCW pumps are located behind the reactor containment building and are mounted on pedestals several feet high. The area is completely exposed to the environment; thus internal flooding could not affect these pumps. Similarly, the ICW pumps are located several hundred feet from the auxiliary building and are also exposed to the outside environment. Both the CCW and ICW pumps are specifically designed for exposure to rain and are housed in NEMA enclosures.

Tank Induced Flooding Sources - A review of available tanks with sufficient water to fill a critical area identified only the RWST, CST, and main condenser as potential flood sources. All other tanks were found to have insufficient volume to be able to fill any of the areas up to the critical damage height. Next the actual flow paths that the water could take from the tanks to affect the areas previously identified were determined. This was done by using the plant's general arrangement drawings and looking for openings, stairwells, and gratings by which the water could spread from room to room. It should be noted that at this step none of the components of interest were under open gratings or openings such that dripping water could lead to damage. Therefore all damage had to occur due to rising water levels. Another item critical to this analysis that was apparent from the drawings and subsequently verified during the plant visit was that the switchgear rooms, auxiliary feedwater pump area, and control and cable spreading room and battery rooms were in physically different building locations. Because of this physical separation, it was difficult for flood waters to originate in one building and be able to affect the other set of components in the other building, especially since there are open areas connecting the buildings. The results of this tank analysis were that the only room that could be flooded out would be the RHR pump room. However, the major tanks of interest are remotely located from these rooms and because the tanks are all outside, and there is a foot high curb into the pump room, in all likelihood the tank floods would not damage the RHR pump room. The rupture of the main condenser would cause the loss of the PCS system; however, the AFW pumps would still be available because they are removed from the main condenser and are bounded by open areas so that water could disperse before it damaged the AFW pumps. In addition, the AFW pumps have a curb to prevent flooding. All other areas are either 1 level above ground level, where all these tanks are located, or are at ground level but have open areas around them where the water could be dispersed. The PCS and Auxiliary feedwater systems were not affected by these tank induced floods. Internal flooding due to tank ruptures at St. Lucie appears not to be a major concern.

Flood Sources found outside the zones of interest - In this portion of the internal flood analysis, the largest flood sources found outside the areas of interest were determined and evaluated as to whether they could lead to enough damage to invalidate all the success paths. The principal sources found outside these areas were:

a. Circulating water for the main condensers
b. Intake cooling water piping
c. Fire main piping

As in the tank case, a rupture of the piping was postulated for this piping on each level of the plant and then flow paths were determined to see if enough of the systems could be damaged. Using the assumptions outlined above, it was found that for a postulated circulating water break at the main condensers (being the largest flood source on the ground level of the plant), that although the PCS system would be lost, for the same reasons as given in the tank analysis, the other critical areas would not be affected. In addition, it was felt that a flood of this nature would cause plant personnel to turn off the circulating water pumps rapidly and therefore there would not be an endless supply of water being pumped into the turbine area, but rather a short gush of water that would then spread out over a large area. (Even if this did not take place, because of the open design at St. Lucie, water would run off into the surrounding canals and streams without any appreciable depth of water being formed in any of the critical areas.)

The other locations in the plant only had ICW and the fire main as principal flood sources on the various plant levels. It was assumed, based on discussions with plant personnel, that an ICW main rupture would cause a reactor trip, but that it would be isolated, and because of the fact that the ICW header at St. Lucie is a loop system with multiple cross connects, the cooling to the systems just mentioned could be restored rapidly. Unless the ICW-induced internal flood caused damage initially to components or their switchgear, they would be available. Because the PCS is not dependent on ICW, it would be available. In addition, ICW runs primarily on the ground level of the plant; thus, the cable spreading room, control room, and battery rooms would not be affected. Furthermore, because of the openness of the plant, the switchgear rooms and AFW pumps would remain undamaged because water could disperse to the open areas. (No ICW runs through the switchgear rooms.

In a similar fashion, the fire mains found at the plant were all smaller than the main service water headers running through the plant and therefore, since the service water headers could not cause sufficient damage, neither could the fire main headers. Note: this refers to fire mains found in areas other than the areas that have been identified as critical.

Largest flood source in each of the areas of interest and potential to affect that zone and other zones - All the areas

0063

of concern have internal flood sources that are the same size as those presented above, or have smaller size sources (typically wet pipe fire suppression systems with two inch diameter pipes) which could not cause damage or have been provided with special protective measures making the possibility of an internal flood in an area remote or have redundant capabilities available elsewhere. The battery rooms fall into the latter case in that they have wet pipe suppression systems, but because there are several of them, redundant capabilities are provided.

Other areas, such as the switchgear rooms and auxiliary feedwater pump area, do not have any fluid sources in their areas. The control rooms do have small flood sources due to a kitchen area that is recessed off the main control room area; however the sink area is not in a direct line of sight with cabinetry so that the question of spray damage is not applicable and the chance of getting six inches of water into the cabinets is deemed incredible from this source. The cable spreading room also has a small flood source from a chilled water cooler; however, the only important safety components in this room are cables, so that spray from this source would have no damaging effects (i.e., cables are not damaged by flood sources). Thus, from this analysis for the electrical distribution and control centers of interest, flood sources within the actual rooms are not capable of causing or leading to a core melt situation.

3.3.3 Potential Modifications

At St. Lucie it appears that none of the plant areas are susceptible to internal flooding or spray. This is primarily due to the fact that the physical layout of the plant is such that almost all potential flood sources are found at ground level, while the ground level arrangement is such that there are many openings to allow the drainage of water. Thus the pooling of water in any significant quantities is remote. Therefore, no modifications are proposed.

3.4 External Flood Analysis

Sources of potential flooding external to a plant are numerous for most plant sites because nearly all plants are located near large bodies of water. River bank overflow, snow melting, heavy precipitation, storm seiches, upstream dam failure, and wave runup are examples of potentially large flood water sources. Flood waters can cause damage or failure to components located in the plant yard as well as damage to equipment inside buildings which may be penetrated by the water. In addition, the buildings themselves may not be designed to withstand the hydrostatic forces of flood waters, resulting in catastrophic structural failures.

However, unlike other natural hazards such as earthquakes and extreme winds, external flooding may not represent a generic hazard to all plant sites since there may be topographic or other physical conditions which preclude site flooding, or reduce the frequency of occurrence. For the St. Lucie Nuclear Power Plant, the uncertainty in design floods was not considered at the time of construction and, therefore the likelihood of extreme floods and their contribution to plant risk has not previously been calculated.

This section will analyze the potential sources of external floods for the St. Lucie Nuclear Power Plant and then quantify the contribution to core melt probability.

A more detailed description of this analysis is provided in Appendix F of this report.

3.4.1 Analysis Methods

It is beyond the scope of the TAP A-45 program to perform an in-depth probabilistic risk assessment. Instead, the objective is to conduct an analysis that quantifies the significant threats to a plant. However, the procedure used to perform this flood analysis embodies the basic philosophy of a full scope probabilistic risk assessment.

To evaluate the probability of plant damage due to external flood events, the following tasks are performed: 1) plant familiarization, 2) hazard analysis, 3) fragility/vulnerability evaluation, 4) systems analysis, and 5) risk quantification. Figure 3.1 shows the relationship of these elements.

The steps in the analysis are summarized below:

Plant Familiarization The purpose of this task is to gather information on the occurrence of external floods hazards at a plant and on the vulnerability of the plant structures and equipment to flooding. Regional studies performed by the Army Corps of Engineers are combined with plant site studies to evaluate the flood frequency. In assessing the reliability of shutdown decay heat removal systems, a review of available documentation (i.e., FSAR, plant drawings and topical reports) and a walkdown of the plant are performed. Safety related components are visually inspected to verify locations and to analyze for possible flooding via all possible pathways.

Hazard Analysis - In the case of external flooding, the hazard analysis is conducted in two steps. First, a screening of the potential sources of flooding (i.e., upstream dams, local precipitation, storm surge, etc.) in the vicinity of the plant is conducted. For sources of flooding that can impact the plant site, an analysis is conducted to evaluate the frequency of occurrence of increasing levels of flood intensity (i.e.,

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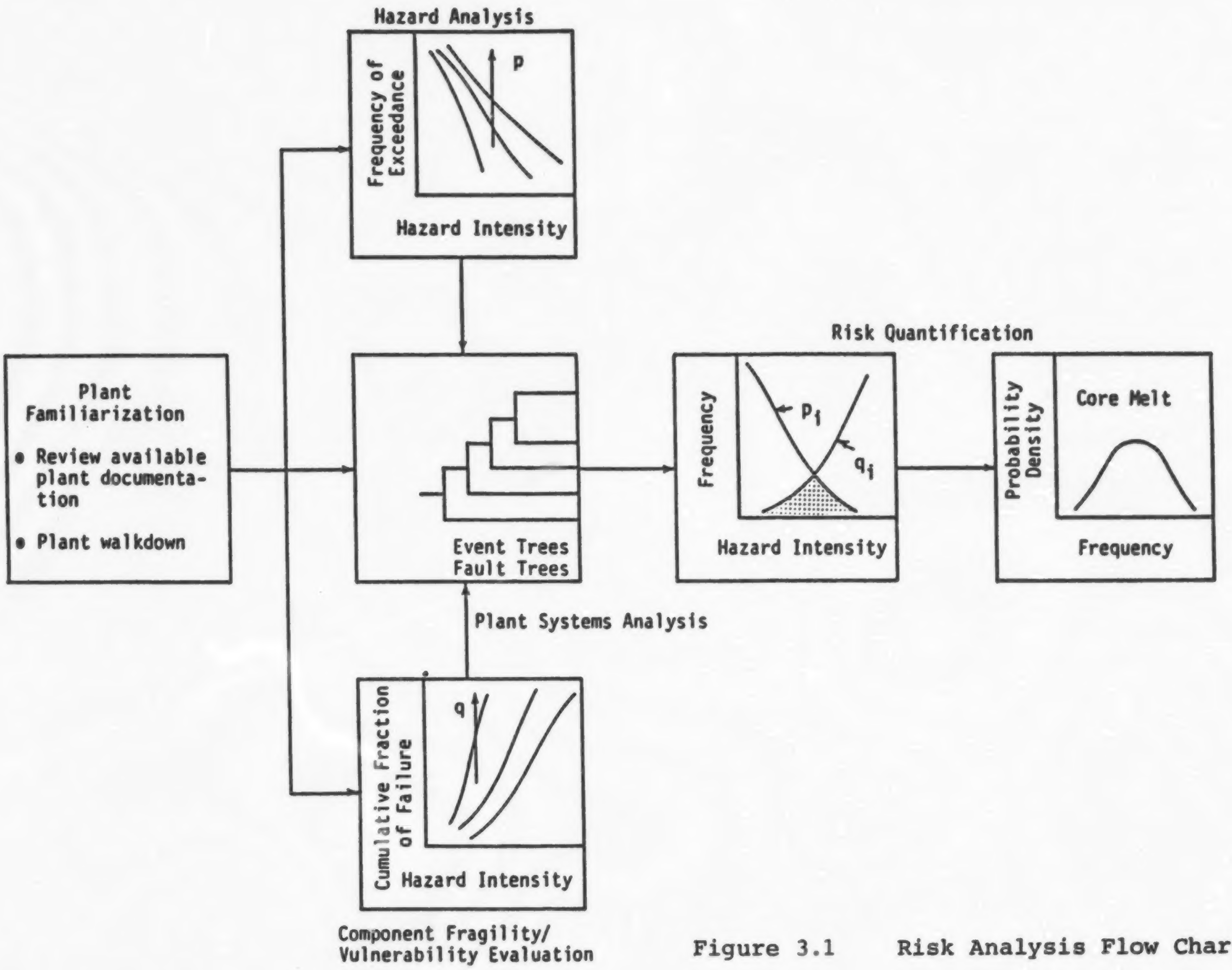
Hazard Analysis
Frequency of Exceedance
p
Hazard Intensity
Plant Familiarization
• Review available plant documentation
• Plant walkdown
Event Trees
Fault Trees
Plant Systems Analysis
Risk Quantification
Frequency
p_i
q_i
Hazard Intensity
Probability Density
Core Melt
Frequency
Cumulative Fraction of Failure
q
Hazard Intensity
Component Fragility/
Vulnerability Evaluation

Figure 3.1 Risk Analysis Flow Chart

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depth of inundation, hydrodynamic forces, etc.). An uncertainty analysis is also performed to assess the modeling or engineering uncertainty in estimating the frequency of occurrence.

Component Fragility Evaluation - For structures and equipment items vulnerable to flood hazards the fraction of failure defined as a function of a flood hazard parameter (i.e., depth of inundation) or fragility curve, is assessed. This is done by 1) identifying the flood protection devices, 2) estimating the depth of submergence necessary to fail a component, 3) determining modes of flooding a room, and 4) transforming local fragility values or critical depth estimates to a site flood hazard characterization. The uncertainties in the modeling used for this evaluation are also considered.

Systems Analysis - The decay heat removal systems analysis is performed by developing event trees and fault trees with an external flooding event as the initiator. The system success criteria are examined for common mode failure due to flooding which could lead to a core melt scenario. The component fragilities are used as input to evaluate the conditional fraction of failure of different safety systems as defined by the systems logic model.

Risk Quantification - The quantification of the likelihood of plant damage is accomplished by properly weighting the conditional fraction of plant damage (i.e., fragility) over the entire range of flood intensities, by their frequency of occurrence. By propagating the uncertainty in the frequency of flood hazards and fragility, the probability distribution on the likelihood of plant damage is evaluated.

3.4.2 External Flood Vulnerabilities

Site Description: The St. Lucie plant is located on Hutchinson Island in St. Lucie County, State of Florida. The plant site is about halfway between Ft. Pierce and Stuart, and approximately 80 miles north of Miami Beach. The Atlantic Ocean serves as the eastern boundary for the site whereas the western boundary is the Indian River, a salt water estuary. Figure 3.2 shows the location of the plant and the topography of the plant site area.

In the vicinity of the plant the terrain is characterized as a low lying salt marsh. The local topography consists of a dune along the Atlantic shore which varies in elevation from about 8 to 14 feet above Mean Low Water (MLW). Between the dunes and the plant there is a salt marsh approximately 1200 feet wide with an average elevation of 2 feet MLW. The salt marsh is bounded on the west by state highway A1A which passes in front of the plant and has a crown elevation of 18.3 feet MLW. Plant structures are founded on engineered fill constructed to an

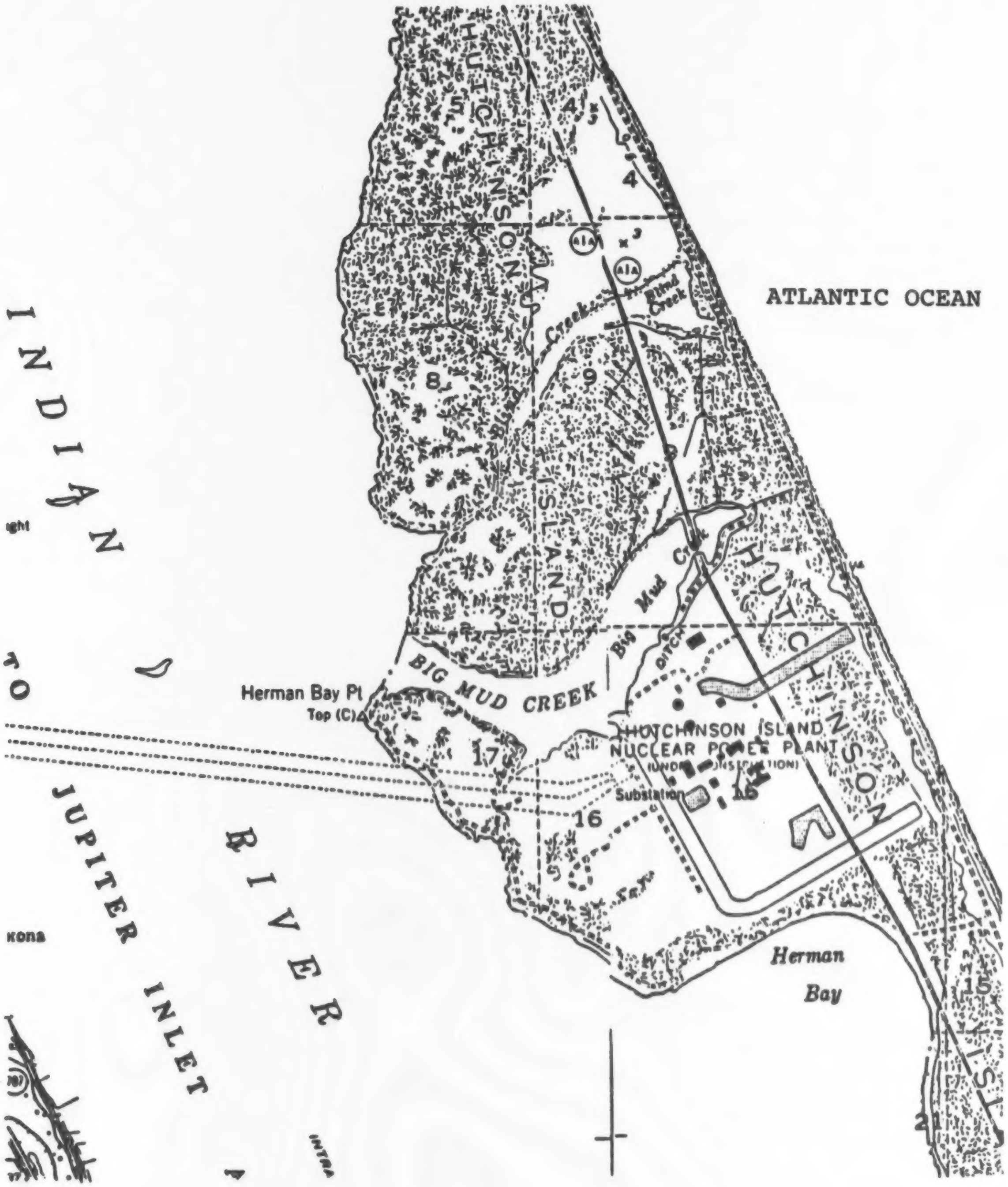
ATLANTIC OCEAN
INDIAN RIVER
TO JUPITER INLET
HUTCHINSON ISLAND
HUTCHINSON ISL
A1A
A1A
Blind Creek
Creek
Big Mud Creek
BIG MUD CREEK
Ditch
Herman Bay Pt
Top (C)
HUTCHINSON ISLAND
NUCLEAR POWER PLANT
(UNDER CONSTRUCTION)
Substation
Herman
Bay
4
4
5
8
9
15
15
16
17
21
ight
Kona
707

Figure 3.2 St. Lucie and Surrounding Area (from USGS Topographic Map)

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elevation of 18 feet above MLW. In the immediate vicinity of plant structures, grade is 19 feet MLW.

Flood Sources: The principal causes of extreme flooding in the vicinity of St. Lucie can be attributed to storm surge and wave action due to tropical storms and hurricanes. During extreme tides and tropical storms or hurricanes, the low lying areas surrounding the plant site are inundated. In addition, rainfall can cause ponding in the low lying areas. However, high soil permeability provides significant ground water recharge from local precipitation, with little runoff. On site, runoff is provided by a storm sewer system and surface runoff to low lying areas and to the intake and discharge canals.

Tidal flooding due to the surge produced by tropical storms has occurred on a number of occasions in the St. Lucie area. Historically, the average frequency of occurrence of tropical storms (including hurricanes) that cross the coast (i.e., landfalling storms only) within 50 miles of St. Lucie is 0.17 events per year. Of these storms, approximately 70% are hurricanes. The frequency of occurrence of exiting storms occurring in proximity to St. Lucie is less than the rate of landfalling storms.

Equipment Location and Protection: Figure 3.3 shows a view of the general arrangement of plant structures and outside equipment items at St. Lucie. Safety-related equipment which is part of the shutdown decay heat removal system is located in the Reactor, Reactor Auxiliary, and Diesel Generator buildings, and on the Intake Structure. In addition, a number of components such as the switchyard, the condensate storage tank, and the diesel fuel oil tanks are located outside.

The St. Lucie plant is designed to withstand the effects of a storm surge that results from the Probable Maximum Hurricane (PMH). The PMH produces a stillwater storm surge of 16.24 feet MLW and wave heights in the discharge canal greater than 10.0 feet. Subsequent to the design of Unit 1, the FSAR for Unit 2 was published. The design basis stillwater surge elevation due to the PMH is 17.2 feet MLW. While wave heights could reach 13.4 feet in the discharge canal, the Unit 2 FSAR concludes that wave runup on the plant site generally does not exceed plant grade of 18.5 feet MLW. However, at the nose of the discharge canal (i.e., the point where the discharge canals for the two units meet) (see Figure 3.4-3), wave runup was calculated to reach 28 feet MLW assuming the embankments were that high. Instead, the runup simply overtops the embankment which has a top elevation of 18.5 feet MLW. According to the Unit 2 FSAR, runup at this location does not affect safety-related equipment since it is more than 400 feet from the nose of the canal. As a result, water which overtops the canal embankment at the nose simply runs off the side slopes,

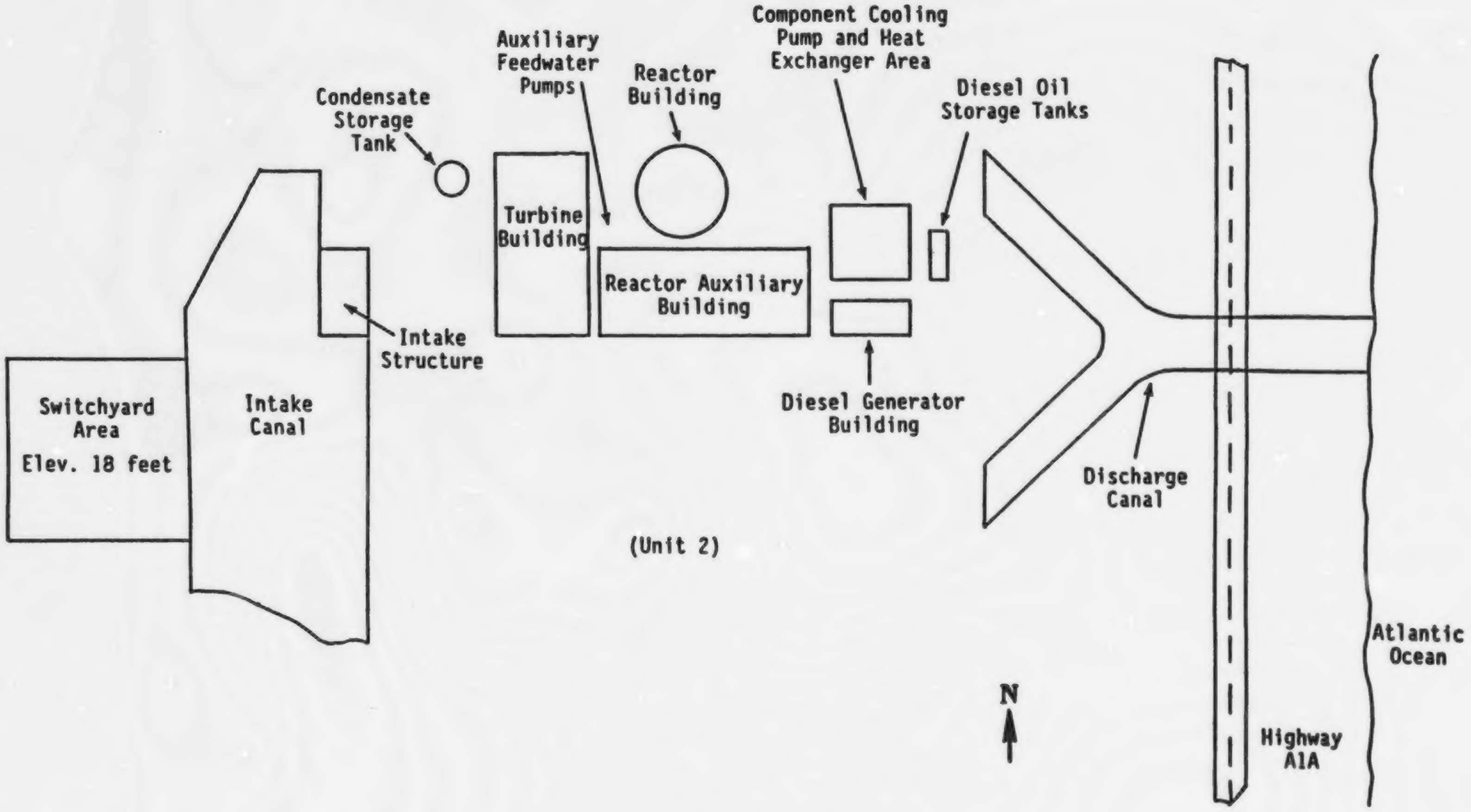
Component Cooling
Pump and Heat
Exchanger Area
Auxiliary
Feedwater
Pumps
Reactor
Building
Diesel Oil
Storage Tanks
Condensate
Storage
Tank
Turbine
Building
Reactor Auxiliary
Building
Intake
Structure
Diesel Generator
Building
Switchyard
Area
Elev. 18 feet
Intake
Canal
Discharge
Canal
(Unit 2)
Atlantic
Ocean
N
Highway
A1A

Figure 3. 3 St. Lucie Plot Plan

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returning to the discharge canal. At the nose of the discharge canal a sheetpile bulkhead is provided to protect against wave runup. The top of the bulkhead reaches elevation 22 feet MLW.

The surge associated with the PMH and accompanying wave action would isolate the plant and result in wave runup above plant grade of 19.0 feet MLW. To mitigate the potential for flooding plant structures and equipment, flood protection measures have been taken.

According to the St. Lucie FSAR, flood protection at the plant is provided to an elevation of at least 19.5 feet MLW. With one exception, all equipment items are protected to elevation 19.5 feet MLW. Also, through an arrangement of plant structures and a floodwall, openings to plant structures are protected from the direct effects of extreme storm surge or wave runup. At the time of the plant visit, emergency procedures for flooding called for stop logs to be inserted at flood wall openings. Thus, complete closure of the flood wall provided protection to elevation 22 feet MLW. However, a subsequent visit by SNL reported that a change in plant emergency procedures indicates that the floodwall is not required, resulting in removal of all stop logs.

At the Intake Structure, safety-related equipment is located above elevation 22 feet MLW, except for the motor operators of isolation valves I-MV-21-2 and I-MV-21-3. Since these components are not required to function during a major flood, their potential vulnerability is not critical.

The two diesel oil storage tanks are protected up to elevation 24.5 feet MLW. Structures that house safety-related equipment are constructed of reinforced concrete and have been conservatively designed to withstand the effects of hydrostatic and buoyant forces associated with a PMH surge elevation of 16.2 feet MLW.

Should flooding occur beyond the floodwall and enter plant structures, safety-related equipment located on the ground floor (elevation 19.5 feet MLW) or at lower elevations is not always protected in watertight rooms.

In the event of a hurricane, emergency procedures are designed to protect plant structures and equipment by closing exterior doors. For a flood which exceeds the height of doorways, 19.5 feet MLW, safety-related equipment could be flooded if leakage occurs.

Based on a review of plant documents and a walkdown of the plant, an assessment of the location and vulnerability of safety-related equipment was made. Table 3.3 lists the vulnerabilities for each of the structures and equipment considered. It was assumed in the development of the

0071

Table 3.3 Vulnerable Structures and Equipment

Structure/Equipment	Location	Comment
Containment Spray Pumps	Located in the Reactor Auxiliary Bldg. at elev. -0.5 ft MLW	Surrounded by interior walls with top elev. +7.5 ft MLW. Doors to room are watertight and kept closed at all times. Room is watertight, therefore components are not vulnerable to external flooding.
High Pressure Safety Injection (HPSI) Pumps	Located in Reactor Auxiliary Bldg. at elev. -0.5 ft MLW	Surrounded by interior walls w/top elev. +7.5 ft MLW. Doors to room are watertight and are kept closed at all times. Room is watertight, therefore components are not vulnerable to external flooding.
Low Pressure Safety Injection (LPSI) Pumps	Located in Reactor Auxiliary Bldg. at elev. -0.5 ft MLW	Surrounded by interior walls w/top elev. +7.5 ft MLW. Doors to room are watertight and kept closed at all times. Room is watertight, therefore components are not vulnerable to external flooding.
Shutdown Heat Exchanger	NW side of Reactor Auxiliary Bldg. at elev. -0.5 ft MLW	Surrounded by interior walls w/top elev. +6.5 ft MLW. Doorways are not watertight. Water can run down the basement if flood level is greater than 19.5 ft MLW.
Auxiliary Feedwater Pumps	Located outside west of Reactor Auxiliary Bldg. and next to Reactor Bldg. at elev. 19.5 ft MLW	Vulnerable to flooding above elev. 19.5 ft MLW.

Table 3.3 Vulnerable Structures and Equipment (cont.)

Structure/Equipment	Location	Comment
Component Cooling Water (CCW) Pumps	Located outside east of Fuel Handling Bldg. in CCW equip. area	Base elev. of equip. is 24 ft MLW. Vulnerable to flooding due to inundation and wave action above 24 ft MLW.
Component Cooling Water Heat Exchangers	Located outside east of Fuel Handling Bldg. in CCW equip. area	Base elev. of equip. is 24 ft MLW. Vulnerable to flooding due to inundation and wave action above 24 ft MLW.
Intake Cooling Water Pumps	Located on west side of plant site on Intake Structure. Pump base elev. is 22 ft MLW	Vulnerable to flood and wave action for elevations greater than 22 ft MLW.
Diesel Oil Storage Tanks (2)	NE of Diesel Generator Bldg., base elev. at 22.5 ft MLW	Flood protection is provided up to 24.5 ft MLW.
Switchyard	Located west of plant at elev. 18' MLW	Vulnerable to inundation and wave action above elev. 18 ft MLW
Diesel Generators	Located in Diesel Generator Bldg. near east side of Reactor Auxiliary Bldg. Operating floor elev. is 22.7 ft MLW.	Vulnerable to inundation above elev. 22.7 ft MLW.
4.16KV Switchgear	Switchgear 1A-2 & 1B-2 located at south end of Turbine Bldg., floor elev. is 19.5 ft MLW.	Vulnerable to inundation above elev. 19.5 ft MLW.
	Switchgear 1A-3 & 1B-3 in Reactor Auxiliary Bldg. at elev. 43 ft MLW	Not vulnerable to flooding.
	Switchgear 1AB located in Reactor Auxiliary Bldg. elev. 28.7 ft MLW	Vulnerable to flooding above elev. 28.7 ft MLW.

Table 3.3 Vulnerable Structures and Equipment (cont.)

Structure/Equipment	Location	Comment
480V Switchgear	Switchgear 1A-1, 1B-1 and AB are on south side of Turbine Bldg. at elev. 19.5 ft MLW.	Vulnerable to flooding above elev. 19.5 ft MLW.
	Switchgear 1A-2 and 1B-2 are in Reactor Auxiliary Bldg. at elev. 43 ft MLW.	Not vulnerable to flooding.
480V Motor Control Centers (MCC)	All 480V MCCs are in Reactor Auxiliary Bldg., elev. 43 ft MLW.	Not vulnerable to flooding.
Batteries	Batteries 1A and 1B are on east side of Reactor Auxiliary Bldg. elev. 44.8 ft MLW.	Not vulnerable to flooding.
Battery Chargers	Battery chargers 1A and 1B are on east side of Reactor Auxiliary Bldg., elev. 43 ft MLW.	Not vulnerable to flooding.
Control Room	Located in north side of Reactor Auxiliary Bldg., elev. 62 ft MLW.	Not vulnerable to flooding.
Remote Shutdown Panel	Located in Reactor Auxiliary Bldg., elev. 43 ft MLW.	Not vulnerable to flooding.

vulnerable structure and equipment list that concrete walls of Category I structures are adequate to protect against hydrostatic or wave forces. In the case of the Reactor Building it is assumed that it is leakproof, and therefore, safety-related components located inside are not vulnerable to flooding.

The switchyard is situated west of the discharge canal and the main plant site, at an elevation of approximately 18 feet MLW. In the event of a severe hurricane which is capable of producing flooding on site, it is anticipated that due to the hurricane-force winds, offsite power will be unavailable.

Precipitation Considerations: As part of the plant design basis, consideration was given to the possibility of flooding due to intense local precipitation. Apparently two analyses were performed. The first analysis was conducted as part of the Unit 1 FSAR. Subsequently, a second analysis was performed as part of the Unit 2 safety analysis. The principal difference in the two studies was the estimate of the Probable Maximum Precipitation (PMP). In the Unit 1 study the 24-hour PMP is estimated to be 32 inches while in the analysis for Unit 2 the PMP is 47.1 inches. In both studies a peak rainfall intensity of 6 inches per hour was considered.

The roofs of structures housing safety-related equipment have been designed to handle rainfall runoff. Measures incorporated in the design include limited parapet wall heights, sloping roofs and drainage by exterior leaders to the storm water drainage system. The site storm drainage system and roof leaders are designed for a rainfall intensity of 6 inches per hour. For more intense rainfall, water could collect and ultimately overtop the roof. The accumulation of water on building roofs does not exceed 2 inches, with the exception of the Reactor Building which has a 1.5 foot parapet wall.

According to the FSAR, ponding which could occur on site does not impact safety-related equipment since the top elevation of any catch basin is anywhere from 6 to 18 inches below the lowest door elevation (19.5 feet MLW). As a result any additional inflow to these areas will overtop the low point in the basin and run off the nuclear island. The FSAR concludes that the extent of local ponding onsite is self limiting and does not impact safety-related equipment or structures.

Risk Quantification: At 19 feet MLW the auxiliary feedwater systems will fail. This will result in the loss of the secondary cooling function. The emergency AC and high pressure injection systems are still available to provide the ECI function via feed and bleed (F&B). The random failure of feed and bleed is conservatively estimated to be 1E-1 at St. Lucie. Therefore, the core melt frequency becomes: (probability the water level exceeds 19 ft MLW but is less than 22 ft MLW) x (failure probability of F&B) = 9.21E-6 x 1E-1 = 9.21E-7.

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At 22 feet MLW the intake cooling water and the component cooling water systems will fail. Therefore, the ECI function will fail in addition to the Secondary Cooling function that fails at 19 feet MLW due to flooding of the AFWS. The frequency of core melt in this case is 2.29E-6. This is just the probability that the water level exceeds 22 ft MLW.

For the above two cases no credit is taken for operator actions to prevent floodings. This is because the procedures are basically designed to prevent leakage but not a major flood. For example, the emergency procedures do not specifically call for the plant to be shut down in the event of a hurricane.

Results: The results of the core melt frequency for the two levels of external flooding at St. Lucie are given in Table 3.4. The largest contribution to core melt is due to a flood level of 22 ft MLW or greater.

Table 3.4 Core Melt Frequency Contributors Due to External Flooding at St. Lucie

Stillwater Flood Level (ft MLW)	Core Melt Frequency
19<x<22	9.21E-7
22<x	2.29E-6
	3.21E-6

It should be noted that for the two flood levels, the recovery of offsite power is not a factor. This is because the safety systems themselves are inundated with water. In addition, the diesel generators are available since they are located at 23.5 feet MLW.

3.4.3 Potential Modifications

Since the total core melt probability contribution is only 3.21E-6, it is not necessary to propose any modifications. However, the existing masonry flood wall appears to no longer have the stop logs available. The first modification which would have been proposed, if external flooding was a dominant contributor, would be to provide the stop logs. These stop logs would prevent flooding of equipment up to a flood level of 22 feet. That is to say that the vulnerability of the plant at the 19 foot level would no longer be a concern. This would reduce the core melt contribution by 9.21E-7 for a minimal cost.

3.5 Extreme Wind Analysis

As part of the Task Action Plan A-45 analysis of the response of decay heat removal systems to internal and external initiating events, a study was performed for wind hazards for the St. Lucie Nuclear Power Plant. Both straight winds of excessive speed and tornado winds with their associated pressure differentials pose potential challenges to specific plant components. In addition, high winds can generate missiles with sufficient impact to penetrate doors, louvers, and metal buildings and severely damage water storage tanks, diesel generators, pumps, and other safety-related systems.

These combined and potentially simultaneous threats of high wind force and wind-generated missiles will be quantified briefly in this section. A more detailed description of the methodology, hazard analysis, component fragilities, and quantification will be presented in Appendix G of this report.

3.5.1 Analysis Methods

To evaluate the probability of plant damage due to external wind events, the following tasks are performed: 1) plant familiarization, 2) tornado and straight wind hazard analysis, 3) tornado missile and wind pressure fragility analyses, 4) systems analysis, and 5) risk quantification. Figure 3.1 shows the interrelationships of these elements.

Plant Familiarization - The purpose of this task is to gather information related to the occurrence of external wind hazards at a plant and the vulnerability of plant structures and equipment to wind effects. In assessing the reliability of decay heat removal systems, a review of available documentation (i.e., FSAR, plant drawings and topical reports) and a walkdown of the plant are performed. Potential missile population at the site, missile pathways, tank locations, equipment exposure, and building design are all visually evaluated for use in later analyses. Previous studies of structural design capacities and regional high wind frequencies are obtained as well as emergency operating procedures.

Hazard Analysis - Mean wind hazard curves for tornadoes and straight winds were developed by the Meteorology and Effluent Treatment Branch of the USNRC for use in the TAP A-45 program. The purpose of the hazard analysis is to develop the uncertainty in the hazard curves. Independent tornado hazard analyses are conducted which include consideration of the uncertainty in the tornado plant strike model, site size, tornado analysis data, tornado damage area and length variation, and Fujita F-scale intensity/wind speed conversion. The resulting curves are scaled so that the resulting mean frequencies of exceedance equal the USNRC values. An analysis of straight winds is conducted to develop the uncertainty in a

0077

similar manner as was done for tornados. Uncertainty in the wind speed data bases, applicability of recording station to the site, and terrain roughness differences are considered.

Tornado Missile and Wind Pressure Fragility Analyses - A tornado missile analysis was performed to develop the conditional probability distribution of missile impact on exposed equipment and on structures containing equipment which could fail due to missile penetration or by secondary impact from spalled concrete. The analysis considers the occurrence frequency of tornados, number of available tornado missiles, target exposure, and target area. Results of past tornado missile impact studies which were based on detailed missile simulations of mass, terminal velocity, deformation characteristics, and angle of impact are used. For structures and equipment vulnerable to wind pressure hazards, the cumulative fraction of failure defined as a function of the windspeed, called a fragility curve, is assessed.

Due to uncertainty, multiple fragility curves are determined for each component.

Systems Analysis - The decay heat removal systems analysis is performed by utilizing the event trees and fault trees developed in the internal analysis to determine which systems are needed to prevent core melt. An event tree is then developed in which wind generated failures are an initiating event with a simultaneous loss of offsite power due to the severity of the storm. The component fragilities and missile impact factors are used to evaluate the conditional probability of failure of different safety systems.

Risk Quantification - The quantification of the likelihood of plant damage is assessed by properly weighting the conditional fraction of plant damage (i.e., fragility) over the entire range of wind intensities by the frequency of occurrence of wind hazards. Recovery actions for each sequence are given credit for where appropriate and, by propagating the uncertainty in the frequency of wind hazards and fragility, the probability distribution of the likelihood of plant damage can be evaluated.

3.5.2 Extreme Wind Vulnerabilities

Vulnerable Structures and Equipment: The safety-related equipment which is part of the shutdown decay heat removal system is contained within the Reactor Building, Reactor Auxiliary Building, and Diesel Generator Building, which are Class I structures. The condensate storage tank (CST), the refueling water storage tank (RWST) and the diesel fuel oil storage tanks are located at ground level outside of the buildings. Finally, the diesel fuel oil transfer pumps, the intake cooling water pumps, and the CCW heat exchanger and pumps are also located outside.

In general, safety-related equipment is protected in Class I structures with exterior concrete walls and roofs at least 12 inches thick. In some locations the walls and roof slabs are thicker. The auxiliary feedwater pumps, intake cooling water pumps and CCW pumps are protected by steel frame structures and 1-inch steel plate or grating.

All Class I structures at St. Lucie were designed as a minimum for a load corresponding to 194 mph windspeed. In addition, the Reactor Building, the Auxiliary Building which encloses the control room, the Fuel Building, the Diesel Generator Building, and all other seismic category I structures needed for the safe shutdown of the plant, are designed for tornado loads stated as follows:

1. A tangential wind velocity of 300 mph.

2. A translational wind velocity of 60 mph (for fuel handling building only).

3. A pressure drop of three psi (except Diesel Generator Building which was designed for 2.25 psi) occurring over a 3-second time interval.

In addition, the structures were designed for the following tornado missiles (note that the tornado pressures were not combined with the missile effects):

1. A 2-inch x 4-inch wooden plank 10 feet long at a velocity of 360 mph.

2. A 4000-pound automobile flying through the air at 50 mph.

Equipment which is not protected and which is vulnerable to wind effects is listed in Table 3.5. It was assumed in the development of the vulnerable structure and equipment list that concrete walls and roof slabs greater than 12 inches thick and heavy steel plates or grating provide adequate protection against tornado missiles.

Although very large missiles could penetrate 1-inch thick steel plates or grating, it is extremely unlikely that large missiles will impact the safety-related equipment located outside. From a probabilistic viewpoint, the contributions to the likelihood of failure of this equipment by missile impact is not significant.

Doors, blockouts, exhaust stacks and other openings to Class 1 structures were postulated to be vulnerable entrances to the buildings for tornado missiles. In general, all openings were either found to be protected by barrier walls, tornado missile barriers or were too small to be significant contributors to the frequency of core melt. Based on a scoping analysis,

Table 3.5 Vulnerable Structures and Equipment

Structure/Equipment	Location	Comment
CCW Pipes	East of Fuel Handling Bldg.	Exposed to tornado missiles.
Diesel Oil Storage Tanks (2)	NE of Diesel Generator Bldg.	Vulnerable to tornado missiles.
RWST	North of Diesel Generator Bldg.	Vulnerable to tornado missiles.
ICW Pipes	West of Turbine Bldg. and East of Fuel Handling Bldg.	Exposed to tornado missiles.

openings less than 100 ft^2 correspond conservatively to a mean frequency of tornado missile impact less than 5.0E-7 per year; thus, the door openings were not found to be significant contributors and were not considered further.

The CST and the diesel oil storage tanks are located outside and are vulnerable to wind effects. These tanks are Class I structures and were not analyzed in this study for wind pressure effects based on experience with similar tanks at other nuclear power plants. Because these tanks are designed for seismic loads, the capacities of the tanks are sufficient to resist wind velocities greater than 360 mph. However, the diesel fuel oil storage tanks and the RWST are vulnerable to tornado missiles and were included in the tornado missile analysis.

The intake cooling water pipes and the component cooling water pipes are not vulnerable to wind pressure but they could be damaged by large tornado missiles and were included in the tornado missile analysis.

Wind Hazard: St. Lucie is located on the east side of Florida on the Atlantic side of the peninsula. The site is located in Region I of the USNRC tornado risk regionalization scheme given in WASH-1300. This is the region which has the highest hazard of the three USNRC regions. The mean values of wind speed frequencies are as follows:

Tornado		Straight Wind	
Wind Speed (mph)	Mean Value	Wind Speed (mph)	Mean Value
66	1.70E-4	80	2.18E-1
93	9.32E-5	90	1.01E-1
134	3.31E-5	111	2.00E-2
182	6.65E-6	120	1.00E-2
234	1.24E-6	128	5.00E-3
290	2.03E-7	138	2.00E-3
349	2.50E-8	144	1.00E-3
		150	5.52E-4
		160	2.65E-4
		174	1.00E-4
		197	2.00E-5
		207	1.00E-5
		230	2.00E-6
		240	1.00E-6

Sequence Quantification: Based on the design and review criteria, all the plant Class I structures exposed to wind were designed for a hurricane wind velocity of 194 mph and the safety-related buildings and structures were designed to resist a tornado of 300 mph tangential wind velocity and a 60 mph translational wind velocity (for fuel handling building only). A wind speed of 300 mph corresponds to a mean frequency of occurrence of 1.5E-7 per year. Based on this result the capacities of the buildings were not considered further. Thus no detailed wind pressure analysis was required for the St. Lucie plant.

In the case of tornado-generated missiles, it was determined that the only credible scenario that could lead to core melt (i.e., failure of both the emergency coolant injection and secondary cooling) is failure of the diesel generators and subsequent battery depletion. One way that this could occur is if missiles were to fail both of the diesel oil storage tanks. The probability of this occurring is the following:

$$P_{\text{core melt}} = (P_{\text{tornado}})*(\text{P missile hits one diesel oil storage tank})*(\text{P missiles hits the other diesel oil storage tank})$$

$$P_{\text{core melt}} = (1.70\text{E-}4)*(2.15\text{E-}3)*(2.15\text{E-}3) = 7.86\text{E-}10$$

The core melt probability due to the failure of one diesel oil storage tank due to a tornado-generated missile, depletion of the related day tank after one hour, and a random failure or test or maintenance unavailability of the redundant diesel generator is calculated to be:

$$P_{\text{core melt}} = (P_{\text{tornado}})*(\text{P missile hits one diesel oil storage tank})*[(\text{P random failure of redundant diesel} + \text{P test or maintenance availability of redundant diesel})]$$

$$\begin{aligned} P_{\text{core melt}} &= (1.70\text{E-}4)*(2.15\text{E-}3)[(3.8\text{E-}2)+(6.1\text{E-}3)] \\ &= 1.61\text{E-}8 \end{aligned}$$

When the additional probability of not recovering offsite AC power before the 8-hour batteries deplete is considered, the core melt probability is indeed negligible (less than 1E-9). In addition, there exist two locked closed cross-connections between the diesel oil storage tanks that would allow fuel to the unaffected diesel generator from the undamaged tank after the day tank is depleted. This would further reduce the core melt contribution due to this accident scenario.

3.5.3 Potential Modifications

The contribution to core melt probability due to extreme winds and/or tornadoes is insignificant. Even though some of the components at St. Lucie were found to be potentially vulnerable to tornado missiles, their failure did not contribute sufficiently to core melt probability to warrant any modifications or additional wind or missile protection.

3.6 Lightning Analysis

Each year at nuclear power plants lighting strikes occur which result in reactor trips, loss of offsite power, and equipment failures. The uncommonly high frequency of this special emergency requires an analysis of the consequences and potential contributions to core melt probability.

This section will present an overview of the methodology for evaluating the vulnerability of a power plant to lightning strike as well as quantification of the resulting probability of core damage for the St. Lucie Nuclear Power Plant.

Further details of analysis methods as well as a compilation of relevant Licensee Event Reports are presented in Appendix H of this report.

3.6.1 Analysis Methods

The evaluation of plant vulnerability to lightning strikes and subsequent determination of potential contribution to core melt probability performed in the TAP A-45 program utilizes the following steps:

a. calculation of the lightning frequency of the site.
b. calculation of the percentage of "damaging" strikes.
c. evaluation of plant lightning protection.
d. evaluation of critical components.
e. evaluation of core melt probability contribution.

The calculation of lightning frequency is determined partially by the number of thunderstorm-days each year for a region and then by the number of cloud-to-ground strikes for a particular site latitude. The number of thunderstorm-days per year will be taken from a figure provided by the U.S. Meteorological Service and included in Appendix H. Several correlations exist to calculate the number of cloud-to-ground strikes for various regions of the United States. For this study, the relationships to be used are:

Location	Ground Flash Density ($km^{-2}yr^{-1}$)
Northern U.S.	0.11T
Southern U.S.	0.17T

where T is the number of thunderstorm-days for the locality of interest. To confirm this correlation and to obtain the range of possible values, a second correlation developed by Pierce will be used as follows:

Cloud-to-ground flash density = (0.1+0.35 sin latitude) x (0.40±0.20)T $km^{-2}yr^{-1}$.

0083

The percentage of these ground strikes which can damage a plant are highly plant specific and dependent upon the adequacy of lightning protection systems. NRC recommendations suggest that plants be designed to withstand a strike of up to 200 kilo-amperes (kA). Since it is beyond the scope of this program to verify the successful operation of these devices, it will be assumed that existing plant lighting protection is adequately designed and maintained. Studies indicate that between one out of one hundred and one out of every thousand ground strikes exceeds 200 kA in current. In this program, we will estimate the probability of a lightning strike being powerful enough to exceed the protective features to be .01. That is, one strike out of every hundred will have a current greater than 200 kA and will be able to damage the plant.

The evaluation of plant protection will be extended only to a literature review, a site visual confirmation, and discussions with plant personnel. The literature review will be limited to studies performed by the plant or information provided in the Final Safety Analysis Report which describe the lightning protection and ground grid system. A site visit is then conducted where the air terminals, earth shield wires and ring conductors are visually confirmed. Additional discussions are held with plant personnel to fill any informational gaps concerning the ground grid system or past lightning occurrences.

The evaluation of critical components is performed in conjunction with the internal analysis of transient initiated events. This approach allows the analyst to identify the success criteria of front line and support systems needed to assure decay heat removal following a reactor trip, loss of offsite power, or other transient which could be caused by a lightning strike. If multiple systems, or multiple trains of a single system, are available for decay heat removal, their sources of water, electric power, instrumentation and control, etc., must be investigated for common mode failures. Again, use of the internal analysis fault tree methodology allows a straight forward approach to this analysis. This approach results in a concentrated list of front line systems and necessary support systems which will prevent core melt if protected against either simultaneous lightning initiated failures, or a lightning strike in conjunction with any number of random system failures.

The calculation of the probability of failure of decay heat removal systems following a lightning strike is the final step in quantifying the contribution to core melt. Given the frequency of lightning striking the site with a current greater than 200 kA, estimates must be made concerning the probability of striking a critical system. If a single system must be hit to lead to core melt, the assumed probability of such a critical hit, given all of the available targets on the plant site, is very conservatively assumed to be one in ten (0.1). If a single component, electrical bus, battery, etc., is the necessary target, the probability of a damaging strike, given all of the available targets, is assumed to be one in one hundred (0.01). However, if a second independent train is available for successful operation, then the probability of either a second lightning strike or a random system failure must be multiplied by the initiating system failure probability. Multiple systems would likewise require multiple lightning strikes or multiple random failures according to the following equation:

$$\text{Core Melt Probability} = T\left[\left(\frac{Ng}{T}\right)(A)(P_{kA})(P_{hit})\right]^{n}(LOSP)(NR)$$

$$+\ (R_1)(T)\left[\left(\frac{Ng}{T}\right)(A)(P_{kA})(P_{hit})\right]^{n-1}(LOSP) + \ .\ .\ .\ . +$$

$$(R_1)(R_2)\ .\ .\ .\ (R_{n-2})(R_{n-1})T\left[\left(\frac{Ng}{T}\right)(A)(P_{kA})(P_{hit})\right](LOSP)(NR)$$

where T = Number of thunderstorm-days per year for the region

Ng = Number of lightning strikes per km^2-year

A = Area of the plant site in km^2

P_{kA} = Probability that the lightning current exceeds the design basis

P_{hit} = Probability that lightning hits a critical component

$LOSP$ = Probability of loss of offsite power

NR = Probability of nonrecovery

n = Number of independent, redundant trains available to prevent core melt

R_n = Probability of random failure of the n_{th} train

3.6.2 Lightning Vulnerability

Using the U.S. Meteorological Services chart of thunderstorm-days in the United States (see Appendix H) for St. Lucie, it is estimated that the site would expect approximately 80 thunderstorm-days each year (T). The correlations for ground flashes (Ng) for the southern U.S. derived by Horn and Ramsey is Ng = 0.17(T). This correlation predicts 13.6 ground flashes $km^{-2}yr^{-1}$ for this southern U.S. site. St. Lucie is located at latitude 27°-20'-58"N which, using the equation derived by Pierce, predicts a ground flash density range of:

$$Ng = (0.1+0.35 \text{ Sin latitude})(0.40 \pm 0.20)T \text{ km}^{-2}\text{yr}^{-1}$$

or $Ng \simeq = 4.2 \rightarrow 12.5/\text{km}^2\text{yr}$.

This is one estimate of the range of ground flash densities for St. Lucie while the value of Ng = 13.6 will be used as a point estimate of the number of strikes per km^2 per year even though it is slightly higher than the upper bound of the range estimate.

It is assumed from the literature review, site inspection, and discussions with plant personnel that the plant was designed according to NRC recommendations that the plant be able to withstand a lightning strike of 200 kA. The probability of exceedance of this current is taken to be 0.01 as seen in Appendix H, Figure H-2. Therefore, the probability of lightning strikes in the St. Lucie region which will exceed the design basis is estimated to be $(0.01)\times(13.6) = 0.136/\text{km}^2\text{yr}$. To simplify calculations, the St. Lucie site will be conservatively estimated to be 1/4 square kilometer in size.

The transient event trees developed for St. Lucie in this program show that either the auxiliary feedwater system or the bleed and feed mode will successfully remove decay heat following a transient. The auxiliary feedwater system consists of multiple trains such that the probability of random failure of all trains is exceedingly small and a single lightning strike cannot fail all of the independent trains. Likewise, the high pressure injection system has multiple trains which are very unlikely to independently fail. However, electric power is needed for the starting and running of pumps, operation of valves, and instrumentation and control. Therefore, the limiting core melt scenario would require a loss of offsite power with simultaneous loss of both DC electric power trains to cause damage.

The Licensee Event Reports (Appendix H) indicate that loss of offsite power is a fairly common occurrence during a powerful thunderstorm. Therefore, it will be assumed that offsite power is lost one half (.5) of the time when lightning storms affect the plant.

Each DC bus can receive power from offsite power, the associated diesel generator, or the batteries dedicated to that train; however, the diesel generator requires the DC batteries in order to start. The scenario where both DC power trains fail due to random or common mode failures is part of the internal analysis of Chapter 2. In this section, the postulated transient involves either two lightning strikes hitting the two DC buses, or a single lightning strike hitting one DC bus while the other DC power train randomly fails. Core melt will commence after about 30 minutes when the steam generators have boiled dry.

These sequence probabilities are calculated as follows:

$$\text{Core Melt Probability} = T\left[\left(\frac{Ng}{T}\right)(A)(P_{kA})(P_{hit})\right]^2(\text{LOSP})(\text{NR}) +$$

$$T\left[\left(\frac{Ng}{T}\right)(A)(P_{kA})(P_{hit})\right](\text{LOSP})(\text{NR})(R_1)$$

where $T = 80$ thunderstorm-days/yr
$Ng = 13.6$ ground flashes/km^{2}yr
$A = 0.25$ km^2
$P_{kA} = 0.01$
$P_{hit} = 0.01$
$\text{LOSP} = 0.5$
NR_{30} min $= 0.5$

It should be noted that the term denoted P_{hit} is a conservative assumption that one out of one hundred lightning strikes greater than 200 kA will damage this particular DC bus. This assumption is probably too high, however, it is beyond the scope of this analysis to trace the pathways of lightning pulses and then assign probabilities to these pathways. The use of a value of 0.01 for P_{hit} will yield an estimate for the damage of a DC bus.

Random failure of a DC power train following a loss of offsite power would require either failure of the bus itself or loss of the power supply from the battery. The failure of the battery will cause failure of the emergency diesel generator to start.

The calculation of the random failure of a DC train of power, then, is as follows:

R_1 = (Bus 1B local fault) + (Bus 1B test or maintenance) + (Battery B local fault) + (Battery B test or maintenance)

The following data is used in this evaluation:

$P_f(\text{BATT})_{LF}$ = 1.2E-3 probability of failure of a battery due to local faults

$P_f(\text{BATT})_{UTM}$ = 1.1E-3 unavailability of a battery due to test or maintenance

$P_f(\text{DCBUS})_{LF}$ = 3E-5 local fault failure probability for a DC bus

$P_f(\text{DCBUS})_{UTM} = \epsilon$

Therefore:

$$R_1 = (3E\text{-}5) + (\epsilon) + (1.2E\text{-}3) + (1.1E\text{-}3) = 2.33E\text{-}3$$

$NR_{30\ min}$ = Failure to recover offsite power in 30 min = 0.5

The probability of core melt is then calculated to be:

$$\text{Core melt} = T\left[\left(\frac{Nq}{T}\right)(A)(P_{kA})(P_{hit})\right]^2(LOSP)(NR) + T\left[\left(\frac{Nq}{T}\right)(A)(P_{kA})(P_{hit})\right](LOSP)(NR)(R_1)$$

$$= 80\left[\left(\frac{13.6}{80}\right)(.25)(.01)(.01)\right]^2(.5)(.5) + 80\left[\left(\frac{13.6}{80}\right)(.25)(.01)(.01)\right](.5)(.5)(2.33E\text{-}3)$$

$$= 4.0E\text{-}10 + 1.98E\text{-}7 = 1.98E\text{-}7/R\ yr$$

For the case where lightning strikes the 125V DC B train and the train A diesel generator fails to start, the A train of DC power will be available for instrumentation and the AFW steam supply valves to the turbine driven pump as well as one feedwater valve could be manually opened. The safety grade batteries are sufficient for eight hours of electric power without recharging.

3.6.3 Potential Modifications

It can readily be seen that all assumptions are conservative in this simplified analysis. Nonetheless, even the worst case lightning scenario does not contribute significantly to the overall core melt probability. It is strongly recommended, however, that auxiliary feedwater valves MV-08-14 and MV-09-12 be modified to receive DC power from 125V DC bus 1A to increase system diversity and eliminate the need for possible manual operation of motor operated valves that could lose train 1B DC power. Currently, the entire turbine driven AFW train is dependent on 125V DC bus 1B.

It was impossible in this program to analyze the subtleties of lightning strikes damaging the control room and then assess the possible consequences. However it would be highly recommended that this building be adequately protected against all lightning strikes according to the Lightning Protection Code NFPA 78. The high incidence of lightning strikes suggests that adequate protection is imperative.

3.7 Sabotage Analysis

Prior studies have suggested that unauthorized activities by insiders (i.e., plant personnel) can be a major source of vulnerabilities. Such acts may include deliberate valve misalignment, improper set point adjustments, physical damage to small diameter cooling lines (e.g., oil and bearing cooling on safety pumps and diesel generators), improper installation of bearings or couplings, and so on. Many such acts, in and of themselves do not cause core melt or a significant radiological release, but given an initiating event such as a loss of offsite power, the failure of these safety components can lead to an increase in public risk. It should be noted that in these investigations the use of explosives by insiders was not considered. This was based upon the earlier studies which suggest that there is such a variety of ways an insider can disable equipment that explosives are not really required. Also, it is presumed that current security measures will prevent the introduction of commercial explosives into the plant.

For purposes of this program, direct overt acts on the part of an outside group were not considered. The evaluation of site security is outside the scope of these investigations. It is presumed here that those plants which meet the requirements of 10 CFR 73.55 are capable of adequately dealing with the external threat. Further, the emphasis on these insider activities was directed toward the potential system vulnerabilities and what can be done to reduce or eliminate them. The analysis took note of existing administrative and security policy and procedures as appropriate, including their potential influence upon insider activities, but it does not examine in detail the effects of changes in such procedures.

Because of the potential sensitivity of detailed discussions of sabotage vulnerabilities, they are not summarized here but are discussed separately in Appendix I. That appendix is issued separately under appropriate SAFEGUARDS controls.

3.8 Summary of Special Emergency Modifications

Table 3.6 presents a compilation of the St. Lucie dominant vulnerabilities for each special emergency analysis performed in this study. The table lists the initiating event and then identifies the vulnerable area or specific component(s). The core melt probability is given both before and after any proposed modification. In the case of the seismic analysis, since the entire plant "sees" the ground acceleration, only a total core melt contribution due to an earthquake is listed.

Sabotage vulnerabilities are not explicitly shown on Table 3.6. However, a number of the special emergency vulnerabilities have sabotage implications, so that sabotage concerns are implicitly included.

Table 3.6 Summary Table of Special Emergency Modifications

Initiating Event	Vulnerable Area/Component	Core Melt Probability/ Rx-yr	Proposed Modification	Modified Core Melt Probability/ Rx-yr
Seismic	RWST + CST Failures	1.33E-5	Provide better tank structures and supports	9.61E-7
Fire	Cable Spreading Room	4.38E-5	Provide a 3M wrap for the B train cable trays	1.46E-5
Internal Flood	None	--	None	--
External Flood	AFW Pumps, ICW Pumps, CCW Pumps	3.21E-6	None	3.21E-6
Extreme Wind	Diesel Oil Storage Tanks	--	None	--
Lightning	DC Bus	1.98E-7	None	1.98E-7

4.0 DEDICATED SHUTDOWN DECAY HEAT REMOVAL SYSTEM

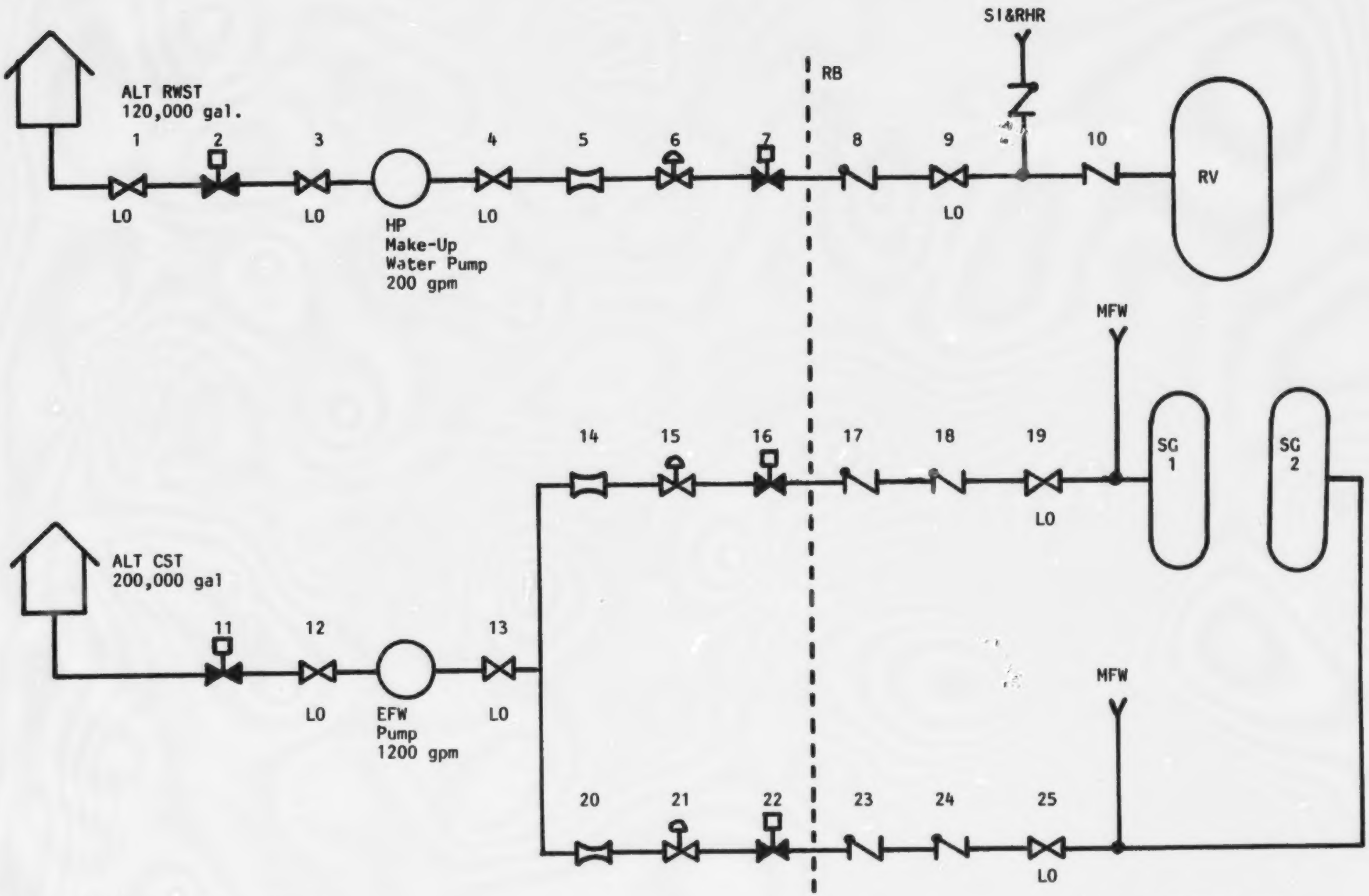
The add-on dedicated shutdown decay heat removal (SDHR) system suggested for Point Beach is a single train of high pressure makeup water and a single train of emergency feedwater with dedicated power and water sources. This type system was selected based upon results from earlier NRC studies (Reference 11) and upon the belief that forced or natural circulation through the steam generators provides the most effective path for decay heat removal from the core. Such a system is also consistent with much of the current European practice (Reference 12).

The SDHR system is a backup system designed to provide emergency core cooling in the unlikely event of a failure of the existing safety systems during a SBLOCA or transient. Core cooling is accomplished by injection of emergency feedwater into the steam generators and release of steam via dedicated atmospheric dump valves. Natural circulation in the reactor coolant loops insures flow and heat removal in the core. The reactor coolant system pressure is maintained above saturation pressure by the use of one group of pressurizer heaters in conjunction with the alternate makeup system. Offsite power will be used when available, otherwise the add-on system is completely independent of the plant systems. The system consists of a high pressure emergency feedwater type pump, a high pressure makeup pump, a storage tank for borated water, a storage tank for feedwater, and the required piping, valving, instruments, and controls for initiation, monitoring and operation of the system. This system is automatically actuated given failure of the primary systems. The structures and connections to the existing system are designed to seismic Category I specifications. A complete description of this system can be found in Reference 11 and in the impact analysis Appendix J. The rationale for a single train system housed in Seismic I structures as contrasted with other options; two-trains, bunkered building, etc., is discussed in Section 6 and 7.

The probabilistic analysis developed here will provide the reliability information for this system required for the value analysis in Section 7. This probabilistic analysis uses the simplified piping diagram shown in Figure 4.1.

The fault tree (Figure 4.2) for the add-on system considers that both the high pressure make up subsystem and the emergency feedwater subsystem are necessary for total system success. Electric power (EP) which is typically attached to each component it supports is brought into the top gate since it is clear that EP is required for system success and the Boolean reduction of the fault tree model with EP on each component it supports would lead to the same result. Room cooling for the pumps is not required. Fans to cool the diesel generator are needed and powered off the diesel generator, but are not modeled explicitly.

0091


ALT RWST
120,000 gal.
1
2
3
4
5
6
7
RB
8
9
SI&RHR
10
RV
LO
LO
LO
LO
HP
Make-Up
Water Pump
200 gpm
MFW
14
15
16
17
18
19
SG
1
SG
2
LO
ALT CST
200,000 gal
11
12
13
LO
LO
EFW
Pump
1200 gpm
MFW
20
21
22
23
24
25
LO

● Point Where Add-On Interfaces Existing Systems

Reference NUREG/CR-2883
Page 2-16, Figure B-95/B-96

Figure 4.1 Simplified Schematic for Dedicated Add-On With Emergency Feedwater and High Pressure Make-Up Water

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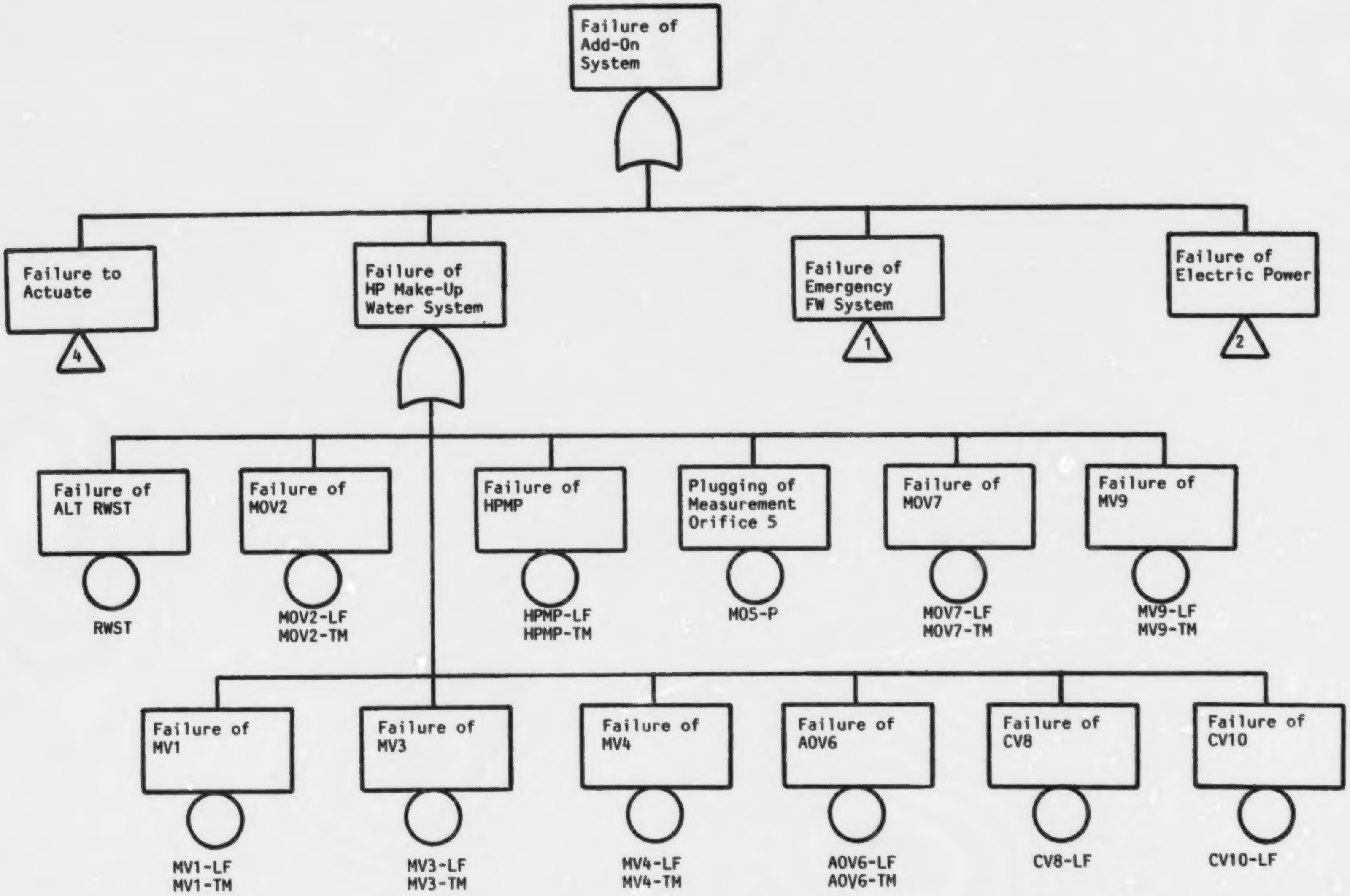
Failure of Add-On System
Failure to Actuate
4
Failure of HP Make-Up Water System
Failure of Emergency FW System
1
Failure of Electric Power
2
Failure of ALT RWST
RWST
Failure of MOV2
MOV2-LF
MOV2-TM
Failure of HPMP
HPMP-LF
HPMP-TM
Plugging of Measurement Orifice 5
MO5-P
Failure of MOV7
MOV7-LF
MOV7-TM
Failure of MV9
MV9-LF
MV9-TM
Failure of MV1
MV1-LF
MV1-TM
Failure of MV3
MV3-LF
MV3-TM
Failure of MV4
MV4-LF
MV4-TM
Failure of AOV6
AOV6-LF
AOV6-TM
Failure of CV8
CV8-LF
Failure of CV10
CV10-LF

Figure 4.2 Potential Add-On SDHR System Fault Tree

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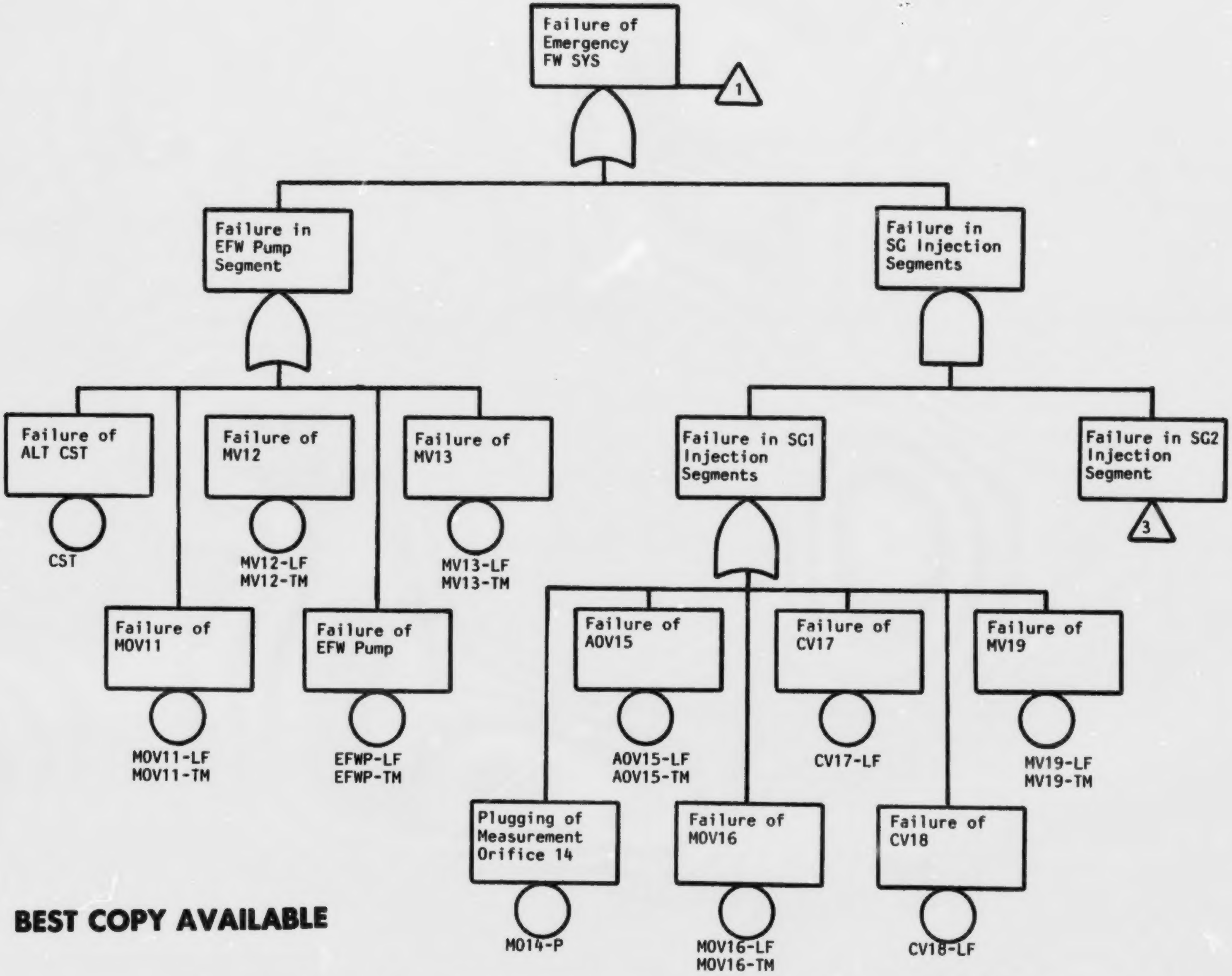
Failure of Emergency FW SYS
1
Failure in EFW Pump Segment
Failure in SG Injection Segments
Failure of ALT CST
CST
Failure of MV12
MV12-LF
MV12-TM
Failure of MV13
MV13-LF
MV13-TM
Failure of MOV11
MOV11-LF
MOV11-TM
Failure of EFW Pump
EFWP-LF
EFWP-TM
Failure in SG1 Injection Segments
Failure in SG2 Injection Segment
3
Failure of AOV15
AOV15-LF
AOV15-TM
Failure of CV17
CV17-LF
Failure of MV19
MV19-LF
MV19-TM
Plugging of Measurement Orifice 14
MO14-P
Failure of MOV16
MOV16-LF
MOV16-TM
Failure of CV18
CV18-LF

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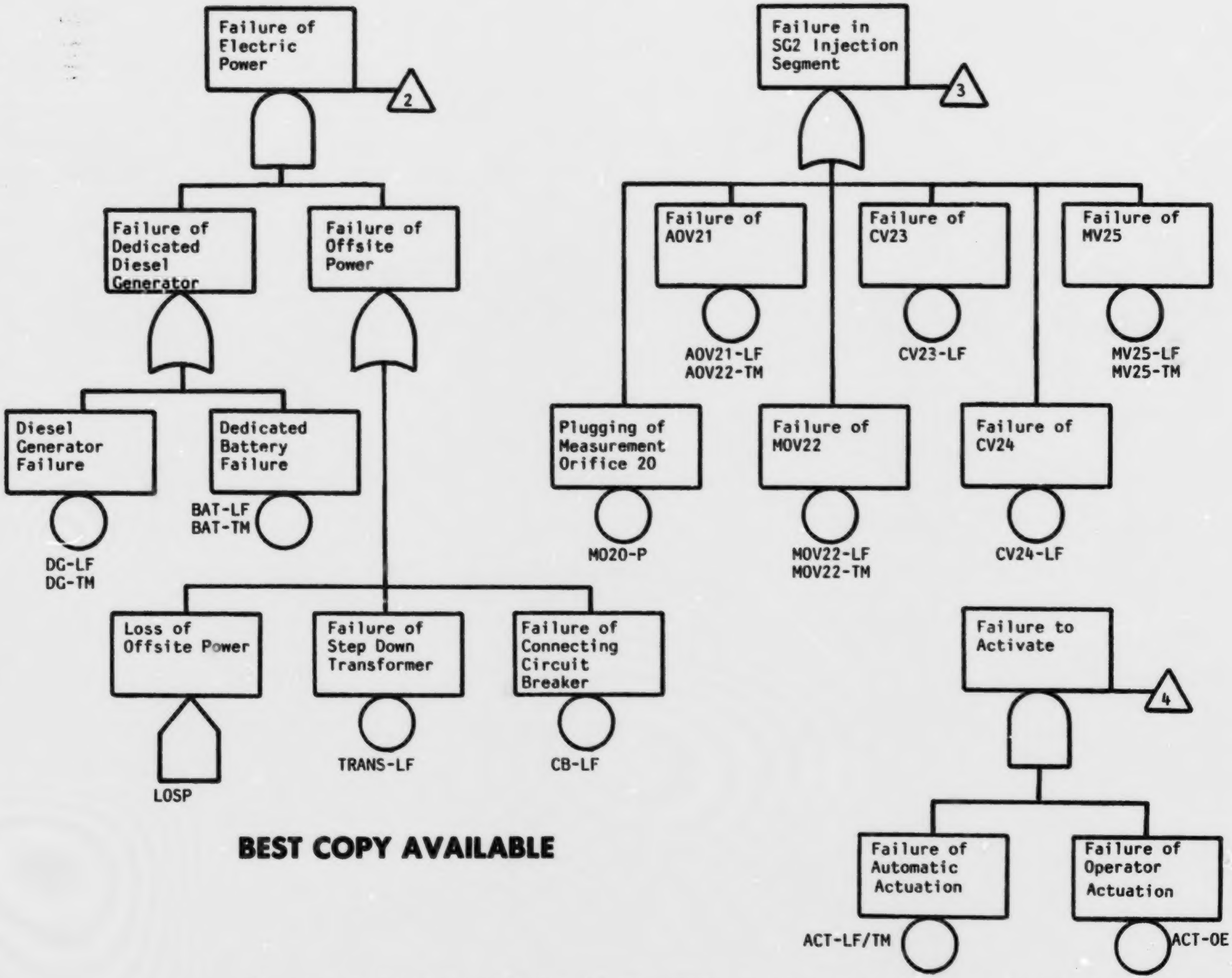
Figure 4.2 Potential Add-On SDHR System Fault Tree (cont.)


Failure of Electric Power
2
Failure of Dedicated Diesel Generator
Failure of Offsite Power
Diesel Generator Failure
DG-LF
DG-TM
Dedicated Battery Failure
BAT-LF
BAT-TM
Loss of Offsite Power
LOSP
Failure of Step Down Transformer
TRANS-LF
Failure of Connecting Circuit Breaker
CB-LF
Failure in SG2 Injection Segment
3
Failure of AOV21
AOV21-LF
AOV22-TM
Failure of CV23
CV23-LF
Failure of MV25
MV25-LF
MV25-TM
Plugging of Measurement Orifice 20
MO20-P
Failure of MOV22
MOV22-LF
MOV22-TM
Failure of CV24
CV24-LF
Failure to Activate
4
Failure of Automatic Actuation
ACT-LF/TM
Failure of Operator Actuation
ACT-OE

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0095

Figure 4.2 Potential Add-On SDHR System Fault Tree (cont.)

needed and powered off the diesel generator, but are not modeled explicitly.

Each of the components shown in the system diagram (Figure 4.1) is depicted in the fault tree (Figure 4.2). If there are multiple failures for the component, these are simply shown below the event box. The acronyms used in the event names are:

Failures:	OE	Operator Error
	LF	Local Fault (hardware)
	TM	Test & Maintenance
	P	Plugging
Event Names:	ACT	Actuation
	RWST	Refueling Water Storage Tank
	CST	Condensate Storage Tank
	MV	Manual Valve
	MOV	Motor Operated Valve
	AOV	Air Operated Valve
	CV	Check Valve
	HPMP	High Pressure Makeup Pump
	EFWP	Emergency Feedwater Pump
	DG	Diesel Generator
	BAT	Battery
	TRANS	Transformer
	CB	Circuit Breaker
	LOSP	Loss of Offsite Power
	MO	Measurement Orifice

Several assumptions were made in the fault tree model, quantification, and subsequent use in the value analysis. These are:

1) All the equipment and the structure containing the equipment are designed to withstand all special emergencies.

2) All the equipment is designed to standard safety equipment specifications.

3) There are no common mode interactions between the redundant discharge legs of the emergency feedwater subsystem.

4) Rupture of the add-on system piping has an insignificant failure probability and thus is not modeled.

5) Offsite power directly from the switchyard will be used to power the add-on system whenever it is available.

6) The system will be automatically actuated with backup actuation by the reactor operator.

Since there are no dependent or common events between branches of the fault tree, the quantification can be accomplished

Table 4.1 Add-On System Value Block

Event	Failure Probability	Source
ACT-OE	3E-3	Ref. 8, Table 20-7, item 3, short list, no checkoff
ACT-LF	5E-3	Ref. 9, Table B10-2
RWST	0	No observed tank failures
CST	0	No observed tank failures
MOX-P	3E-4	Ref. 10, Table 5.1-1, item 4.4.1
HPMP-TM	2.1E-3	Ref. 8, Table 20-7, item 1, failure to restore after T&M (1E-3)(1/12 monthly check) plus 2E-3 due to downtime
EFWP-TM	2.1E-3	Ditto
HPMP-LF	9E-3	Ref. 10, Table 5.1-1, item 1.1 (3E-3/d) + (3E-5/h) x 24 hrs + (5E-3) control circuit failures
EFMP-LF	9E-3	Ditto
CVX-LF	1E-4	Ref. 10, Table 5.1-1, item 2.4.1
MVX-LF	1E-4	Ref. 10, Table 5.1-1, item 2.6.2 (1/2) x(3E-7/h)x(720 hrs) monthly test interval
AOVX-LF	4E-5	Ditto except 1E-7/h from item 2.1.2
MOVX-LF	8E-3	Ref. 10, Table 5.1-1, item 2.1.1, failure to open on demand (3E-3) plus control circuit failures (5E-3)
MVX-TM	3.8E-4	Ref. 8, Table 20-7, item 1, failure to restore after T&M (1E-3)(1/12 monthly check) plus (3E-4) due to downtime
AOVX-TM	3.8E-4	Ditto
MOVX-TM	3.8E-4	Ditto
DG-TM	6.1E-3	Ditto except 6E-3 due to downtime
BAT-TM	1.1E-3	Ditto except 1E-3 due to downtime
DG-LF	3.8E-2	See Appendix B3.3.4
BAT-LF	3.9E-3	See Appendix B3.3.1
TRANS-LF	2.4E-5	Ref. 10, Table 5.1-1, item 4.5.1 (1E-6/h) x 24 hrs
CB-LF	3E-5	Ref. 10, Table 5.1-1, item 4.1.2
LOSP	1	For LOSP initiated transients
	0	For all other initiating events since the failure frequency of LOSPs already included as 1E-3
MOV-CM	4E-4	Ref. Appendix B, Section 3.3.4

X ≡ variable depending on the component number

0097

associated probability; for example, ACT-OE/3E-3 means that 3E-3 is the probability of the event ACT-OE, actuation operator error.

Failure of the Add-on System = the sum of

(ACT-OE/3E-3 * ACT-LF/5E-3) + RWST/0 + MV1-LF/1E-4 +
MV1-TM/3.8E-4 + MOV2-LF/3E-3 + MOV2-TM/3.8E-4 + MV3-LF/1E-4 +
MV3-TM/3.8E-4 + HPMP-LF/9E-3 + HPMP-TM/2.1E-3 + MV4-LF/1E-4 +
MV4-TM/3.8E-4 + MO5-P/3E-4 + AOV6-LF/4E-5 + AOV6-TM/3.8E-4 +
MOV7-LF/8E-3 + MOV7-TM/3.8E-4 + CV8-LF/1E-4 + MV9-LF/1E-4 +
MV9-TM/3.8E-4 + CV10-LF/1E-4 + CST/0 + MOV11-LF/8E-3 +
MOV11-TM/3.8E-4 + MV12-LF/1E-4 + MV12-TM/3.8E-4 + EFWP-LF/9E-3 +
EFWP-TM/2.1E-3 + MV13-LF/1E-4 + MV13-TM/3.8E-4 + (MO14-P/3E-4 +
AOV15-LF/4E-5 + AOV15-TM/3.8E-4 + MOV16-LF/8E-3 +
MOV16-TM/3.8E-4 + CV17-LF/1E-4 + CV18-LF/1E-4 + MV19-LF/1E-4 +
MV19-TM/3.8E-4) * (MO20-P/3E-4 + AOV21-LF/4E-5 +
AOV21-TM/3.8E-4 + MOV22-LF/8E-3 + MOV22-TM/3.8E-4 +
CV23-LF/1E-4 + CV24-LF/1E-4 + MV25LF/1E-4 + MV25-TM/3.8E-4) +
(DG-LF/3.8E-2 + DG-TM/6.1E-3 + BAT-LF/3.9E-3 + BAT-TM/1.1E-3) *
(LOSP + TRANS-LF/2.4E-5 + CB-LF/3E-5)+MOV-CM

After the probabilities for the basic events are summed the equation reduces to three terms.

(5.12E-2)+(9.78E-3)*(9.78E-3)+(4.91E-2)*(LOSP+5.4E-5)+4E-4

When the terms are multiplied out and combined, the equation further reduces to:

(5.17E-2)+(4.91E-2)*LOSP

This indicates that the add-on system failure is dominated by seven events in the first term (i.e., MOV2-LF, HPMP-LF, HPMP-TM, MOV7-LF, MOV11-LF, EFWP-LF, and EFWP-TM) and LOSP with DG-LF, DG-TM, and BAT-LF. In other words, the probability of failure lies in the operating components and electric power. Finally, considering the two possible conditions for availability of electric power, the system failure probabilities become:

p(add-on system failure given offsite power is available)
= 5.2E-2 per demand

and

p(add-on system failure given offsite power is unavailable)
= 1.0E-1 per demand.

These values will be used in the value analysis outlined in Section 7.

5.0 ALTERNATIVE SELECTION AND INTEGRATION

Fourteen individual potential vulnerabilities were identified during the initial internal and special emergency core melt probability analyses for the St. Lucie Nuclear Plant. These vulnerabilities are given in Table 5.1 with the approximate core melt frequency attributed to each of them. Even though the vulnerabilities and associated modifications described here were identified in a probabilistic type analysis, the diversity of the modifications is such that several potential difficulties that might be envisioned are covered. Furthermore, as has been the experience in other programs, it is often the support systems, not the front line systems, which cause problems. Sabotage vulnerabilities are not explicitly shown in Table 5.1, but it should be noted that the special emergency vulnerabilities have sabotage implications so that it is implicitly included here. All of these vulnerabilities and the corresponding modifications are discussed in more detail in the appropriate subsections of Sections 2.0 and 3.0. Combinations of modifications will become the basis for the alternatives.

In the early stages of the program during the safety criteria screening process, a number of insights regarding potential vulnerabilities were identified. These insights are listed in Table 5.2. Some of these early insights correlate with the vulnerabilities found in the subsequent analyses. These are:

Initial Insights from Table 5.2	Internal and Special Emergency Analysis Vulnerabilities from Table 5.1
Item 2	Seismic Vulnerability 1
Item 4	Internal Vulnerability 1
	Internal Vulnerability 4
	Internal Vulnerability 5
	Internal Vulnerability 6
	Internal Vulnerability 8
Item 5	Fire Vulnerability 1

To a degree this tends to validate the initial qualitative screening process.

While modifications were suggested for many of the vulnerabilities identified, only 3 modifications were carried into the impact analysis. Each of these modifications addresses a specific vulnerability. The dedicated SDHR system described in Section 4.0, however, applies to all of the vulnerabilities, internal and external, including sabotage. Combinations of modifications for specific vulnerabilities and/or the dedicated SDHR system are called alternatives. Two alternatives were selected as shown in Table 5.3.

Table 5.1

Potential St. Lucie Vulnerabilities

Vulnerabilities	Approximate Contribution to Core Melt Probability of Sequences Involving This Vulnerability
Fire 1 - Cable Spreading Room Fire	4.4E-5
Seismic 1 - RWST and CST Failures	1.3E-5
Flood 1 - Surge Floods Safety Systems	3.2E-6
Internal 1 - Common Mode Battery Failure	2.9E-6
Internal 2 - CCW Pump Common Mode Failure	1.4E-6
Internal 3 - ECC Recirculation Motor Operated Valves Local Failures	8.5E-7
Internal 4 - Diesel Generator Common Mode Failure and AFW Turbine Driven Pump Local Fault	8.2E-7
Internal 5 - Local Diesel Generator Faults and AFW Turbine Driven Pump Local Fault	8.0E-7
Internal 6 - Diesel Generator 1 Local Fault and Battery B Local Fault	4.5E-7
Internal 7 - ICW Pump Common Mode Failure	3.5E-7
Internal 8 - Diesel Generator 1 Local Fault and Battery B Test or Maintenance Outage	2.7E-7
Lightning 1 - Lightning Strikes DC Bus	2.0E-7
Internal 9 - SIS Logic Test or Maintenance Common Mode Failure	1.5E-7
Internal 10 - RAS Logic Test or Maintenance Common Mode Failure	7.0E-8

Table 5.2

Initial St. Lucie Insights

1. There is only one steam inlet valve for the AFW turbine pump which accepts steam from two steam generators.
2. There is only one water source normally valved open to the AFW pumps.
3. The RHR shutdown cooling heat exchangers have a valve on either side of the parallel exchangers which can stop flow through the RHR.
4. The onsite emergency power system consists of one pair of diesel generators and one pair of batteries.
5. Remote shutdown capability is for hot shutdown only, and assumes main control room is "not damaged."

Table 5.3

St. Lucie Potential Alternatives

Modification:	Alternatives 1	 2
Fire 1	X	
Seismic 1	X	
Dedicated SDHR Add-on		X

All the Xs in a column combine into an alternative.

0101

As noted above, these vulnerabilities were identified on the basis of core melt probability, but their scope is broad enough to include nonquantified events also. The total core melt probability for those initiating events covered in this program is greater than the sum of the individual vulnerability contributions due to the residual core melt probability that is attributable to internal accident sequences and cut sets that were not dominant and to special emergency events that were not quantified. The base case core melt probability is discussed in detail in Section 7.1.1. The maximum potential value that can be obtained by a modification is the core melt probability of the vulnerabilities it addresses, i.e., to have a perfect modification which reduces that vulnerability to zero.

The alternatives in Table 5.3 were selected to obtain a variety of potential benefits. Table 5.4 gives the preliminary estimates used in establishing the alternatives. In this stage of the assessment, a reliability factor of 0.1 was used for the dedicated SDHR system based upon earlier work.[11] In Section 7.1.2 the failure rates with and without offsite power were used for the appropriate sequences. For purposes of the alternative selection the individual modifications were assumed to improve (reduce) the core melt contribution of any particular event by a factor of 3E-2. This assumption was based on engineering judgment. The actual core melt probability improvement established by analysis after the conceptual design was complete is used in the value discussions in Section 7.1.2.

Table 5.4

Preliminary Estimated Core Melt Probabilities Given Modifications Applied to St. Lucie

Total Core Melt Probability Before Modifications	7.4E-5
Alternative 1	1.9E-5
Alternative 2	7.4E-6

Alternative 1 was selected to address the most important external vulnerabilities that had been identified. This alternative combines two modifications to reduce the vulnerabilities to fire and seismic which, based upon the above assumptions, improved the core melt probability by a factor of about 4. Alternative 2 considers the addition of a dedicated shutdown decay heat removal system by itself. It is assumed that this system not only improves the core melt probability (a factor of about 10) but also addresses certain unquantifiable vulnerabilities (e.g., sabotage) and aspects of residual risk.

In summary, the alternatives were defined assuming that individual modifications would reduce the contribution to core

0102

melt probability by a factor of 3E-2 where applied. This provided only one such alternative--one solely for the external event sequences of fire and seismic. The dedicated shutdown decay heat removal system addresses all vulnerabilities and, alone, constitutes alternative 2. The predicted overall reduction of core melt probability ranged from a factor of about 4 to a factor of about 10. The impact analysis for these alternatives is discussed in Section 6 and the anticipated reductions in core melt probability based on the analysis of the conceptual designs are presented in Section 7.

6.0 ALTERNATIVE IMPACT ANALYSIS

The impact analysis addresses the two alternatives identified in Section 5.0 which consist of two potential plant modifications including a dedicated shutdown decay heat removal system.

6.1 Methodology and Approach

6.1.1 Objectives

The objectives of the impact assessment were to determine all the potential impacts resulting from the implementation of each alternative. These are:

1) Cost of equipment and installation of the modifications included in a specific alternative in 1985/1986 dollars,

2) Cost of replacement power, if required, in order to implement an alternative in 1985/1986 dollars,

3) Cost of subsequent annual operation, maintenance, and in-service inspections that would be associated with the alternative in 1985/1986 dollars, and

4) Radiation exposure involved in the installation of the alternative in person-rem.

It was also desired to obtain potential yearly changes to occupational radiation exposure due to the alternatives and replacement power costs if the alternatives would affect the normal outage times. These two impacts could be positive or negative theoretically, however, they were both judged to be negligible for the St. Lucie Plant.

6.1.2 Approach

Given the vulnerability identification and proposed alternatives, initial design reports were prepared on each modification. These reports described the proposed design, interfaces with other systems and structures, the scope of work to be performed such as bill of materials and rip-out required, and major construction activities. A site inspection plan was prepared for each modification which identified all the specific areas to be visited and questions to be asked in order to properly estimate the impact of the alternatives. The preliminary design was done based upon drawings such as building layouts and piping and instrumentation drawings that were obtained prior to the plant visit and the experience of the architectural engineer from previous similar activities.

The next step was a plant visit to verify the design and to collect further information to estimate costs relative to such

0104

factors as congestion, access, feasibility of equipment locations, and radiation levels. In practice the plant visit is essential and at St. Lucie numerous improvements were made to the proposed modifications during the visit. Details that would be effectively impossible to obtain from available written material were obtained by observation and discussions with knowledgeable plant personnel.

Using the information collected during the site visit to improve the initial design, a final design was accomplished for each modification. Costs and radiation exposures associated with implementing the designs were then estimated using standard industry practices to produce the impact analysis. Details of the design and costs are given in Appendix J. A brief summary statement of the content of each alternative is given in Table 6.1.

6.2 Results

The results for the St. Lucie Plant are based on a set of economic ground rules and assumptions shown in Table 6.2 for both generic and local factors such as labor rates, replacement power costs per day, remaining economic life of the plant, owner's costs, and contingencies. The details of these factors are described in Appendix J. However, one example is owner's costs which consider factors such as retraining operators, health physics, quality assurance, review of contractor design, interaction with the NRC, and preparation of procedure and manual changes. Much of the detail available in Appendix J is provided for completeness of the impact estimates and possible use in any follow-on analyses. The basic information needed for the value-impact analysis in Section 9.0 is summarized in Table 6.3 using the local basis costs which is consistent with the use of local values in the value impact analysis.

The value-impact analysis variables shown in Table 6.3 (i.e., I_1, I_3, and V_3) reflect the fact that occupational dose is treated as a negative value rather than as a positive impact. Furthermore, the yearly occupational radiation dose (V_4) is also treated as a negative value but is considered to be negligible in the estimates for the two St. Lucie alternatives. Similarly, the yearly replacement power costs (I_4) for installation of the modifications are considered to be negligible for St. Lucie. Thus, Table 6.3 contains the necessary information to perform the value-impact analysis.

Although the local cost basis was used in the value-impact analysis, it is appropriate to note the ratio of the generic to local engineering and installation costs which are:

Alternative	Generic/Local Cost Ratio
1	1.14
2	1.11

These ratios show that the generic costs are about 14% greater for the alternative including only modifications to the existing plant and about 11% greater for the alternative involving the add-on dedicated shutdown decay heat removal system. (NOTE: If the values were computed for a generic site there would also be variations, but it cannot be stated a priori whether they would be greater or less.)

Table 6.1 Content of the Proposed Alternatives

Alternative 1 combines the following two modifications:

a. Cable Spreading Room Fire Protection System

This modification reduces plant vulnerability to a fire in the cable spreading room by enclosing B train cable tray that is essential to safety in an intumescent sheet that provides an excellent heat barrier to retard fire transmission. A fire in the cable spreading room will therefore be prevented from spreading and becoming a common mode failure for decay heat removal systems.

b. Condensate Storage and Refueling Water Storage Tank Seismic Supports

This modification increases the seismic strength of both the Unit 1 CST and RWST by adding additional supports. For the CST, supports were designed between the tank and the surrounding reinforced missile shield wall. For the RWST, 4 inch wide vertical flanges are welded to the bottom 10 feet of the tanks at approximately 8 places around the tank. An additional horizontal brace is welded to the vertical flanges about 5 feet above the ground to prevent buckling during an earthquake. These fixes will ensure a source of water for the auxiliary feedwater pumps as well as the high pressure, low pressure, and containment spray injection systems.

ALTERNATIVE 2 - Alternative Decay Heat Removal Add-on System

The purpose of this modification is to improve the reliability of the decay heat removal function by adding additional, protected redundancy. It involves the addition of a new, complete, and dedicated decay heat removal system to assure core cooling and reactor coolant makeup following a loss-of-coolant accident or external events such as fire, flood, tornado, and earthquake. One of its subsystems will supply emergency feedwater to two of the plant's steam generators. Decay heat would be removed by releasing steam through new steam generator atmospheric steam dump valves. The other of its subsystems provides makeup to the Reactor coolant system in order to maintain inventory. Both of these subsystems are powered from a dedicated diesel and fed from a dedicated water supply.

This "Add-on" Decay Heat Removal System (ADHR) is housed in two new seismic Category I buildings designed to resist tornado missiles. The ADHR building contains the following components: Alternate Decay Heat Removal Pump, Alternate Service Air Compressor, Alternate Diesel Generator and Associated Tanks, Alternate Makeup Pump, heating and

Table 6.1 Content of the Proposed Alternatives (cont.)

ventilating units, valves, instruments and controls. The other new building contains the Borated Water Storage Tank, Alternate Condensate Storage Tank, Tank Circulation Pumps, valves, instruments and controls.

The Alternate Decay Heat Removal Pump takes suction from the Alternate Condensate Storage Tank and discharges into the existing auxiliary feedwater piping outside containment. The Alternate Makeup Pump takes suction from the new Borated Water Storage Tank and discharges to the safety injection header outside Reactor Containment and then to the Reactor Vessel.

0108

Table 6.2
Economic Ground Rules and Assumptions
From Appendix J

	PARAMETER	GENERIC	LOCAL
1.	Plant Location	Eastern Pennsylvania	Southern Florida
2.	Remaining Economic Life (yrs)	19 yrs	24 yrs
3.	Inflation Rate (%)	6	6
4.	Discount Rate (%)	10.5	10
5.	Fixed Charge Rate (%)	17.0	16.5
6.	O&M Levelization Factor	1.47	1.63
7.	Replacement Power Costs ($/MW(e) - Day)	550 (1)	584
8.	Average Refueling Shutdown Length (Days)		
	PWR	60	60
	BWR	70	--
9.	O&M Labor Rate ($/Hr) (2)	22.80	20.61
10.	O&M Labor/Materials Ratio	40/60	40/60
11.	Owner's Costs (%)	10	10
12.	Contingency (%)	25	25

Note: (1) MAAC Power Pool Average

(2) Includes employer's social security and workmen's compensation contributions

0109

Table 6.3 St. Lucie Impact Analysis Results

Alternative	Engineering and Installation Costs in 1985 Dollars \$ x 10^{-3}	Operations and Maintenance Cost in 1985 Dollars \$ x 10^{-3}	Occupational Dose During Installation in Person-Rem*
1	881	15	0
2	53791	348	20
Value-Impact Analysis Variable	I_1	I_3	V_3

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* The occupational dose for alternative 1 is substantially less than that for alternative 2 because there is no need to enter containment to make the modifications.

7.0 ALTERNATIVE VALUE ANALYSES

This section will describe the core melt probabilities and public risk estimates for the base case before any suggested modifications are applied and the improvements resulting from each alternative, i.e. combination of modifications.

7.1 Core Melt Probabilities

7.1.1 Base Case Probabilities

The core melt probability is the sum of the probabilities from the internal analysis and each of the special emergency analyses except for sabotage which is as yet not clearly quantifiable. The following list gives each probability and the section of this report where it was derived.

Core Melt Probability		Reference Section
Internal	1.4E-5	App. B Sec. 6.1.2, Sec. 2.2
Seismic	1.3E-5	App. C Sec. 7.0, Sec. 3.1.2
Internal Flood	ε	App. E Sec. 6, Sec. 3.3.2
Fire	4.4E-5	App. D Sec. 7, Sec. 3.2.2
Wind & Missiles	ε	App. G Sec. 7.2.2, Sec. 3.5.2
External Flood	3.2E-6	App. F Sec. 6.2, Sec. 3.4.2
Lightning	2.0E-7	Sec. 3.6.2

7.48E-5 base core melt probability

7.1.2 Alternative Probabilities

The alternative probabilities are determined from the combination of internal and special emergency modifications included in that alternative. The value analysis for each modification is found in the applicable appendix.

The special emergency and internal analyses are essentially independent in the St. Lucie value analysis (there are impact dependencies). This is not a generic situation, but it is generally the case for St. Lucie. Each of the alternatives will now be discussed and evaluated.

Alternative 1 Core Melt Probability - Alternative 1 is the combination of the modifications for fire and for seismic. The core melt probability for alternative 1 is the sum of the internal and external core melt probabilities with the above modifications.

Internal (no modifications)	1.44E-5
Seismic (modifified)	9.61E-7
Internal Flood (no modification)	ε
Fire (modified)	1.46E-5
Wind and missiles (no modification)	ε
External flood (no modification)	3.21E-6
Lightning (no modification)	1.98E-7

p(CM,alt 1) = 3.34E-5

Alternative 2 Core Melt Probability - Alternative 2 is the addition of a dedicated shutdown decay heat removal system as described in Section 4.0. As noted in Section 4, the reduction in core melt probability from the add-on system changes depending upon whether or not offsite power is available. Also, because the add-on is an independent system, the improvement in core melt probability is simply the product of the add-on system unreliability and the base case probability. The add-on system unreliability is 5.2E-2 if offsite power (OSP) is available and 9.8E-2 if offsite power is not available. Therefore, to properly apply the add-on system unreliability the core melt probability for internal events must be split into that when OSP is available and that when OSP is not available. In special emergencies from flood, wind, and lightning it is assumed offsite power is not available. For fire initiated events, offsite power is generally available, while for seismic the availability of offsite power varies with the magnitude of the earthquake. In the internal analysis situation the core melt probability is partitioned by the accident sequence applicability to offsite power availability. Alternative 2 with the add-on applied to the base case is as follows:

	Base Case Probability	Add-on Unreliability	Alt 2 Core Melt Probability
Internal - OSP available	5.84E-6	5.2E-2	3.04E-7
Internal - OSP not available	8.52E-6	9.8E-2	8.35E-7
Seismic - OSP not available	9.89E-6	9.8E-2	9.69E-7
Seismic - OSP available	3.42E-6	5.2E-2	1.78E-7
Fire - OSP available	4.38E-5	5.2E-2	2.28E-6
Wind - OSP not available	ε	9.8E-2	ε
External Flood - OSP not available	3.21E-6	9.8E-2	3.15E-7
Lightning - OSP not available	1.98E-7	9.8E-2	1.83E-8

Thus p(CM,alt 2) = 4.90E-6

7.1.3 Summary of Probabilistic Core Melt Estimates

The results of the two alternatives can be summarized by the core melt probabilities and change in core melt probability from the base value.

Alternative	Probability of Core Melt	Change in Probability of Core Melt from Base Case	Factor of Improvement from Base Case Probability
Base case	7.48E-5		
1	3.34E-5	4.14E-5	2.2
2	4.90E-6	6.99E-5	15.3

The effect of an alternative can be viewed in several ways. One way is to compare the alternative core melt probability to some desired probability. A second approach is simply to examine the relative improvement over some base situation. For example, alternative 1 is approximately a factor of 2 improvement and alternative 2 is approximately a factor of 15 improvement from the base value. Further conclusions will be discussed in Sections 9.0 and 11.0.

7.2 Public Risk Estimates

In past PRAs several public risk measures were used (e.g., early deaths, early injuries, latent deaths, property damage). The measure used in this analysis is the averted offsite dose resulting from the release of radioactive material to the atmosphere and subsequent dispersion out to a radius of 50 miles from the plant site. All other values (positive or negative) are considered in the impact analysis (Section 6.0).

The averted offsite dose in dollars is calculated for the internal analysis and each special emergency.

7.2.1 Base Case Estimates

The base case estimate of the public risk measure (population dose) is computed from the base case probabilities for each accident sequence including containment systems, the containment failure mode probabilities and release category assignments, and the consequences for each release category. The base case internal analysis accident sequences with containment systems are found in Table B.28. The base case special emergency sequence probabilities are summarized from the various appendices in Table 7.1. The containment failure mode (CFM) probabilities and release category assignments for each type containment systems sequence are summarized in Table 7.2.

0113

Table 7.1 St. Lucie Special Emergency Accident Sequence Probabilities Including Containment Systems

Special Emergency	Containment Systems Sequence	Base Case Probability	Probability After Modification
Flood	$\bar{Y}_1\bar{C}_1\bar{F}'$	1.15E-6	NA
Flood	Y_1C_1	2.29E-6	NA
Fire	Y_1C_1	4.38E-5	1.46E-5
Lightning	Y_1C_1	1.98E-7	NA
Wind	Y_1C_1	ε	NA
Seismic	$\bar{Y}_1\bar{C}_1\bar{F}$	1.28E-7	4.47E-8
Seismic	$\bar{Y}_1C_1$	1.26E-5	ε
Seismic	Y_1C_1	4.69E-7	8.89E-7

0114

Table 7.2 St. Lucie Accident Sequence to Release Category Mapping

Containment Systems Sequence*	Special Conditions	Containment Failure Mode with Probability and Release Category				
	Core Melt	α	β	γ,δ_e	δ_ℓ	ε
$\bar{Y}_1\bar{C}_1\bar{F}'$		1E-4	2E-3	1.4E-2	1.8E-1	2.5E-1
		1	5	3	5	7
$\bar{Y}_1\bar{C}_1F'$	EM	1E-4	2E-3	1.4E-2	1.8E-1	2.5E-1
		1	5	3	4	6
	LM	1E-4	2E-3	1.4E-2	1.8E-1	2.5E-1
		1	4	2	3	6
$\bar{Y}_1C_1$		1E-4	2E-3	1.4E-2	1.8E-1	2.5E-1
		1	4	2	3	6
$Y_1\bar{C}_1\bar{F}'$		1E-4	2E-3	1.4E-2	1.8E-1	2.5E-1
		1	5	3	5	7
$Y_1\bar{C}_1F'$	EM	1E-4	2E-3	1.4E-2	1.8E-1	2.5E-1
		1	5	3	4	6
	LM	1E-4	2E-3	1.4E-2	1.8E-1	2.5E-1
		1	4	2	3	6
Y_1C_1		1E-4	2E-3	1.4E-2	1.8E-1	2.5E-1
		1	4	2	3	6

*The exact definition of the containment system sequence varies for the PWRs depending upon the system success criteria.

EM ≡ early core melt, LM ≡ late core melt.

The results are tabulated in Table 7.3 and shown by dominant accident sequence in Table 7.4 in a format similar to that used in past PRAs, e.g., WASH 1400. The release category (RC) probabilities are then combined with the consequences determined in Appendix K to obtain the expected population dose in person-rem per reactor-year and cost in dollars (at $1000 per person-rem) in Table 7.5. Three values are presented in Table 7.5: an upperbound for purposes of this study which is based upon WASH 1400 type source terms, our central estimate which we have defined as 0.3 times the WASH 1400 source term, and a lower bound which is defined as 0.1 times the WASH 1400 source term. This selection of source terms should not be interpreted as an endorsement of any particular set. The "real" source term may be larger or smaller. These were selected in order to provide some indication of the sensitivity of these results to variations in the source term.

Two examples may be useful to show how the calculation was done. First, consider the internal accident sequence $S_2MH_1'H_2'$. This is sequence number 1 found in Table B.28 where it is seen that only the $\bar{Y}_1\bar{C}_1F'$ containment sequence is significant. Its probability is 1.0E-6, i.e.,

$$p(S_2MH_1'H_2- \bar{Y}_1\bar{C}_1F') = 1.0E-6.$$

This is a late melt sequence resulting from the ECR failure (i.e., $H_1'H_2'$). Going to Table 7.2 this falls in the third row of the table where 5 containment failure modes (CFM) are possible. As an illustration consider the most likely CFM which is ε with a probability of 2.5E-1 and a release category 6 (RC6) assignment. Thus,

$$p(S_2MH_1'H_2' - \bar{Y}_1\bar{C}_1F' - \varepsilon) = (1.0E-6)*(2.5E-1) = 2.5E-7.$$

This sequence is placed in row 1 (S_2 initiating event) column 6 (RC6) in Table 7.4. The initiating event segregation in Table 7.4 is not important for any reason except perhaps to make it easy to locate a sequence. This same procedure is repeated for every significant internal accident sequence.

Next consider a seismic special emergency sequence with the associated containment systems sequence Y_1C_1. Its probability from Table 7.1 is 4.69E-7, i.e.,

$p(\text{SEISMIC} - Y_1C_1) = 4.69E-7,$

In this case let us consider all the CFMs. These are,

$p(\text{SEISMIC}-Y_1C_1 - \alpha) = (4.69E-7)*(1E-4) = 4.69E-11$ RC1,

0116

Table 7.3 St. Lucie - Base Case

Accident Sequence	Total Prob.	RC 1		RC 2		RC 3		RC 4		RC 5		RC 6		RC 7	
		Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.
Internal-$\bar{Y}_1\bar{C}_1\bar{F}'$	1.81E-6	α	1.81E-10			γ,δ_e	2.53E-8			β δ_ℓ	3.62E-9 3.26E-7			ε	4.53E-7
Internal-$\bar{Y}_1\bar{C}_1F'$(LM)	1.85E-6	α	1.85E-10	γ,δ_e	2.59E-8	δ_ℓ	3.33E-7	β	3.70E-9			ε	4.63E-7		
Internal-$\bar{Y}_1C_1$	6.25E-7	α	6.25E-11	γ,δ_e	8.75E-9	δ_ℓ	1.13E-7	β	1.25E-9			ε	1.56E-7		
Internal-$Y_1\bar{C}_1\bar{F}'$	5.40E-11	α	5.40E-15			γ,δ_e	7.56E-13			β δ_ℓ	1.08E-13 9.72E-12			ε	1.35E-11
Internal-$Y_1\bar{C}_1F'$(EM)	4.96E-9	α	4.96E-13			γ,δ_e	6.94E-11	δ_ℓ	8.93E-10	β	9.92E-12	ε	1.24E-9		
Internal-Y_1C_1	8.43E-6	α	8.43E-10	γ,δ_e	1.18E-7	δ_ℓ	1.52E-6	β	1.69E-8			ε	2.11E-6		
Lightning-Y_1C_1	1.98E-7	α	1.98E-11	γ,δ_e	2.77E-9	δ_ℓ	3.56E-8	β	3.96E-10			ε	4.95E-8		
Flood-$\bar{Y}_1\bar{C}_1\bar{F}'$	9.21E-7	α	9.21E-11			γ,δ_e	1.29E-8			β δ_ℓ	1.84E-9 1.67E-7			ε	2.30E-7
Flood-Y_1C_1	2.29E-6	α	2.29E-10	γ,δ_e	3.21E-8	δ_ℓ	4.12E-7	β	4.58E-9			ε	5.73E-7		
Fire-Y_1C_1	4.38E-5	α	4.38E-9	γ,δ_e	6.13E-7	δ_ℓ	7.88E-6	β	8.76E-8			ε	1.10E-5		
Seismic-$\bar{Y}_1\bar{C}_1\bar{F}'$	1.28E-7	α	1.28E-11			γ,δ_e	1.79E-9			β δ_ℓ	2.56E-10 2.30E-8			ε	3.20E-8
Seismic-$\bar{Y}_1C_1$	1.26E-5	α	1.26E-9	γ,δ_e	1.76E-7	δ_ℓ	2.27E-6	β	2.52E-8			ε	3.15E-6		
Seismic-Y_1C_1	4.69E-7	α	4.69E-11	γ,δ_e	6.57E-9	δ_ℓ	8.44E-8	β	9.38E-10			ε	1.17E-7		
TOTALS			7.31E-9		9.83E-7		1.27E-5		1.41E-7		5.22E-7		1.76E-5		7.15E-7

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Table 7.3 St. Lucie - Base Case - Alternative 1

Accident Sequence	Total Prob.	RC 1		RC 2		RC 3		RC 4		RC 5		RC 6		RC 7	
		Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.
Internal-$\bar{Y}_1\bar{C}_1\bar{F}'$	1.81E-6	α	1.81E-10			γ,δ_e	2.53E-8			β δ_ℓ	3.62E-9 3.26E-7			ϵ	4.53E-7
Internal-$\bar{Y}_1\bar{C}_1F'$(LM)	1.85E-6	α	1.85E-10	γ,δ_e	2.59E-8	δ_ℓ	3.33E-7	β	3.70E-9			ϵ	4.63E-7		
Internal-$\bar{Y}_1C_1$	6.25E-7	α	6.25E-11	γ,δ_e	8.75E-9	δ_ℓ	1.13E-7	β	1.25E-9			ϵ	1.56E-7		
Internal-$Y_1\bar{C}_1\bar{F}'$	5.40E-11	α	5.40E-15			γ_e,δ	7.56E-13			β δ_ℓ	1.08E-13 9.72E-12			ϵ	1.35E-11
Internal-$Y_1\bar{C}_1F'$(EM)	4.96E-9	α	4.96E-13			γ,δ_e	6.94E-11	δ_ℓ	8.93E-10	β	9.92E-12	ϵ	1.24E-9		
Internal-Y_1C_1	8.43E-6	α	8.43E-10	γ,δ_e	1.18E-7	δ_ℓ	1.52E-6	β	1.69E-8			ϵ	2.11E-6		
Lightning-Y_1C_1	1.98E-7	α	1.98E-11	γ,δ_e	2.77E-9	δ_ℓ	3.56E-8	β	3.96E-10			ϵ	4.95E-8		
Flood-$\bar{Y}_1\bar{C}_1\bar{F}'$	9.21E-7	α	9.21E-11			γ,δ_e	1.29E-8			β δ_ℓ	1.84E-9 1.67E-7			ϵ	2.30E-7
Flood-Y_1C_1	2.29E-6	α	2.29E-10	γ,δ_e	3.21E-8	δ_ℓ	4.12E-7	β	4.58E-9			ϵ	5.73E-7		
Fire-Y_1C_1	1.46E-5	α	1.46E-9	γ,δ_e	2.04E-7	δ_ℓ	2.63E-6	β	2.92E-8			ϵ	3.65E-6		
Seismic-$\bar{Y}_1\bar{C}_1\bar{F}'$	4.47E-8	α	4.47E-12			γ,δ_e	6.26E-10			β δ_ℓ	8.94E-11 8.05E-9			ϵ	1.12E-8
Seismic-$\bar{Y}_1C_1$	8.89E-7	α	8.89E-11	γ,δ_e	1.24E-8	δ_ℓ	1.60E-7	β	1.78E-9			ϵ	2.22E-7		
TOTALS			3.17E-9		4.04E-7		5.24E-6		5.87E-8		5.07E-7		7.22E-6		6.94E-7

0118

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Table 7.3 St. Lucie - Base Case - Alternative 2

Accident Sequence	Total Prob.	RC 1		RC 2		RC 3		RC 4		RC 5		RC 6		RC 7	
		Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.	Mode	Prob.
Internal-$\bar{Y}_1\bar{C}_1\bar{F}'$	1.63E-7	α	1.63E-11			γ,δ_e	2.24E-9			β δ_ℓ	3.26E-10 2.93E-8			ε	4.08E-8
Internal-$\bar{Y}_1\bar{C}_1F'$(LM)	1.81E-7	α	1.81E-11	γ,δ_e	2.53E-9	δ_ℓ	3.26E-8	β	3.62E-10			ε	4.53E-8		
Internal-$\bar{Y}_1C_1$	6.13E-8	α	6.13E-12	γ,δ_e	8.58E-10	δ_ℓ	1.10E-8	β	1.23E-10			ε	1.53E-8		
Internal-$Y_1\bar{C}_1\bar{F}'$	5.29E-12	α	5.29E-16			γ,δ_e	7.41E-14			β δ_ℓ	1.06E-14 9.52E-13			ε	1.32E-12
Internal-$Y_1\bar{C}_1F'$(EM)	2.88E-10	α	2.88E-14			γ,δ_e	4.03E-12	δ_ℓ	5.18E-11	β	5.76E-13	ε	7.20E-11		
Internal-Y_1C_1	5.33E-7	α	5.33E-11	γ,δ_e	7.46E-9	δ_ℓ	9.59E-8	β	1.07E-9			ε	1.33E-7		
Lightning-Y_1C_1	1.94E-8	α	1.94E-12	γ,δ_e	2.72E-10	δ_ℓ	3.49E-9	β	3.88E-11			ε	4.85E-9		
Flood-$\bar{Y}_1\bar{C}_1\bar{F}'$	9.03E-8	α	9.03E-12			γ,δ_e	1.26E-9			β δ_ℓ	1.81E-10 1.63E-8			ε	2.26E-8
Flood-Y_1C_1	2.24E-7	α	2.24E-11	γ,δ_e	3.14E-9	δ_ℓ	4.03E-8	β	4.48E-10			ε	5.60E-8		
Fire-Y_1C_1	2.28E-6	α	2.28E-10	γ,δ_e	3.19E-8	δ_ℓ	4.10E-7	β	4.56E-9			ε	5.70E-7		
Seismic-$\bar{Y}_1\bar{C}_1\bar{F}'$	1.10E-8	α	1.10E-12			γ,δ_e	1.54E-10			β δ_ℓ	2.20E-11 1.98E-9			ε	2.75E-9
Seismic-$\bar{Y}_1C_1$	1.09E-6	α	1.09E-10	γ,δ_e	1.53E-8	δ_ℓ	1.96E-7	β	2.18E-9			ε	2.73E-7		
Seismic-Y_1C_1	4.59E-8	α	4.59E-12	γ,δ_e	6.43E-10	δ_ℓ	8.26E-9	β	9.18E-11			ε	1.15E-8		
TOTALS			4.70E-10		6.21E-8		8.01E-7		8.93E-9		4.81E-8		1.11E-6		6.62E-8

0119

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Table 7.4 St. Lucie Dominant Accident Sequences and Special Emergencies Contributing to Release Categories

		RELEASE CATEGORIES						
		1	2	3	4	5	6	7
INITIATING EVENTS	S_2 LOCA	$S_2MH'_1H'_2-F'-\alpha$ 1.0E-10 $S_2MD_1D_1-C_1-\alpha$ 4.2E-11 $S_2MD_1D_2-Y_1C_1-\alpha$ 1.2E-10	$S_2MH'_1H'_2-F'-\gamma,\delta_e$ 1.4E-8 $S_2MD_1D_2-C_1-\gamma,\delta_e$ 5.9E-9 $S_2MD_1D_2-Y_1C_1-\gamma,\delta_e$ 1.7E-8	$S_2MD_1D_2-Y_1C_1-\delta_\ell$ 2.2E-7 $S_2MH'_1H'_2-F'-\delta_\ell$ 1.8E-7 $S_2MD_1D_2-C_1-\delta_\ell$ 7.6E-8	$S_2MH'_1H'_2-F'-\beta$ 2.0E-9 $S_2MD_1D_2-C_1-\beta$ 8.4E-10 $S_2MD_1D_2-Y_1C_1-\beta$ 2.4E-9		$S_2MH'_1H'_2-F'-\epsilon$ 2.5E-7 $S_2MD_1D_2-C_1-\epsilon$ 1.1E-7 $S_2MD_1D_2-Y_1C_1-\epsilon$ 3.0E-7	
	T_1 TRANSIENT	$T_1MLE-\alpha$ 1.5E-10 $T_1MLE-Y_1F'-\alpha$ 4.3E-13 $T_1MLE-Y_1C_1-\alpha$ 5.9E-10 $T_1MQD_1D_2-Y_1C_1-\alpha$ 1.9E-11 $T_1MQLD_1-Y_1C_1-\alpha$ 2.9E-11	$T_1MLE-Y_1C_1-\gamma,\delta_e$ 8.3E-8 $T_1MQD_1D_2-Y_1C_1-\gamma,\delta_e$ 2.7E-9 $T_1MQLD_1-Y_1C_1-\gamma,\delta_e$ 4.1E-9	$T_1MLE-\gamma,\delta_e$ 2.1E-8 $T_1MLE-y_1f'\gamma,\delta_e$ 6.0E-11 $T_1MLE-Y_1C_1-\delta_\ell$ 1.1E-6 $T_1MQD_1D_2-Y_1C_1-\delta_\ell$ 3.4E-8 $T_1MQLD_1-Y_1C_1-\delta_\ell$ 5.2E-8	$T_1MLE-Y_1F'-\delta_\ell$ 7.7E-10 $T_1MLE-Y_1C_1-\beta$ 1.2E-8 $T_1MQD_1D_2-Y_1C_1-\beta$ 3.8E-10 $T_1MQLD_1-Y_1C_1-\beta$ 5.8E-10	$T_1MLE-\beta$ 3.0E-9 $T_1MLE-\delta_\ell$ 2.7E-7 $T_1MLE-Y_1F'-\beta$ 8.6E-12	$T_1MLE-Y_1F'-\epsilon$ 1.1E-9 $T_1MLE-Y_1c_1-\epsilon$ 1.5E-6 $T_1MQD_1D_2-Y_1C_1-\epsilon$ 4.8E-8 $T_1MQLD_1-Y_1C_1-\epsilon$ 7.3E-8	$T_1MLE-\epsilon$ 3.8E-7
	T_2 TRANSIENT	$T_2MLE-\alpha$ 1.2E-11 $T_2MQH'_1H'_2-F'-\alpha$ 1.0E-11 $T_2MQD_1D_2-C_1-\alpha$ 2.5E-12	$T_1MQH'_1H'_2-F'-\gamma,\delta_e$ 1.4E-9 $T_2MQD_1D_2-C_1-\gamma,\delta_e$ 3.5E-10	$T_2MLE\ -\gamma,\delta_e$ 1.7E-9 $T_2MQH'_1H'_2-F'-\delta_\ell$ 1.8E-8 $T_2MQD_1D_1-C_1-\delta_\ell$ 4.5E-9	$T_2MQH'_1H'_2-F'-\beta$ 2.0E-10 $T_2MQD_1D_1-C_1-\beta$ 5.0E-11	$T_2MLE-\beta$ 2.4E-10 $T_2MLE-\delta_\ell$ 2.2E-8	$T_2MQH'_1H'_2-F'-\epsilon$ 2.5E-7 $T_2MQD_1D_2-C_1-\epsilon$ 6.3E-9	$T_2MLE-\epsilon$ 3.0E-8
	$T_{3,4,5}$ TRANSIENTS	$T_3QH'_1H'_2-F'-\alpha$ 7.5E-11 $T_3QD_1D_1-C_1-\alpha$ 1.8E-11 $T_3QD_1D_2-Y_1C_1-\alpha$ 8.5E-11 $T_4MLE-\alpha$ 1.7E-11 $T_4MLE-Y_1-\alpha$ 5.4E-15 $T_4MLE-Y_1F'-\alpha$ 6.6E-14 $T_5MLE-\alpha$ 1.9E-12	$T_3QH'_1H'_2-F'-\gamma,\delta_e$ 1.1E-8 $T_3QD_1D_2-C_1-\gamma,\delta_e$ 2.5E-9 $T_3QD_1D_2-Y_1C_1-\gamma,\delta_e$ 1.2E-8	$T_3QD_1D_1-Y_1C_1-\delta_\ell$ 1.5E-7 $T_4MLE-\gamma,\delta_e$ 2.4E-9 $T_4MLE-Y_1-\gamma,\delta_e$ 7.6E-13 $T_4MLE-Y_1F'-\gamma,\delta_e$ 9.2E-12 $T_5MLE-\gamma,\delta_e$ 2.7E-10 $T_3QH'_1H'_2-F'-\delta_\ell$ 1.4E-7 $T_3QD_1D_2-C_1-\delta_\ell$ 3.2E-8	$T_3QH'_1H'_2-F'-\beta$ 1.5E-9 $T_3QD_1D_2-C_1-\beta$ 3.6E-10 $T_3QD_1D_2-Y_1C_1-\beta$ 1.7E-9 $T_4MLE-Y_1F'-\delta_\ell$ 1.2E-10	$T_4MLE-\beta$ 3.4E-10 $T_4MLE-Y_1-\beta$ 1.1E-13 $T_5MLE-\beta$ 3.8E-11 $T_4MLE-\delta_\ell$ 3.1E-8 $T_4MLE-Y_1-\delta_\ell$ 9.7E-12 $T_4MLE-Y_1F'-\beta$ 1.3E-12 $T_5MLE-\delta_\ell$ 3.4E-9	$T_3QH'_1H'_2-F'-\epsilon$ 1.9E-7 $T_3QD_1D_2-C_1-\epsilon$ 4.5E-8 $T_3QD_1D_2-Y_1C_1-\epsilon$ 2.1E-7 $T_4MLE-Y_1F'-\epsilon$ 1.7E-10	$T_4MLE-\epsilon$ 4.3E-8 $T_5MLE-\epsilon$ 4.8E-9 $T_4MLE-Y_1-\epsilon$ 1.4E-11

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0120

Table 7.4 St. Lucie Dominant Accident Sequences and Special Emergencies Contributing to Release Categories (cont.)

		RELEASE CATEGORIES						
		1	2	3	4	5	6	7
INITIATING EVENTS	SPECIAL EMERGENCIES	Lightning-Y_1C_1-α 2.0E-11	Lightning-Y_1C_1-γ,δ_e 2.8E-9	Lightning-Y_1C_1-δ_ℓ 3.6E-8	Lightning -Y_1C_1-β 4.0E-10	Seismic-β 2.6E-10	Lightning-Y_1C_1-ε_ℓ 5.0E-8	Flood-ε_ℓ 2.3E-7
		Flood-α 9.2E-11	Flood-Y_1C_1-γ,δ_e 3.2E-8	Flood-γ,δ_e 1.3E-8	Flood-Y_1C_1-β 4.6E-9	Seismic-δ_ℓ 2.3E-8	Flood-Y_1C_1-ε 5.8E-7	Seismic-ε 3.2E-8
		Flood-Y_1C_1-α 2.3E-10	Fire-Y_1C_1-γ,δ_e 6.1E-7	Flood-Y_1C_1-δ_ℓ 4.1E-7	Fire-Y_1C_1-β 8.8E-8	Flood-β 1.8E-9	Fire-Y_1C_1-ε 1.1E-5	
		Fire-Y_1C_1-α 4.4E-9	Seismic-C_1-γ,δ_e 1.8E-7	Fire-Y_1C_1-δ_ℓ 7.9E-6	Seismic-C_1-β 2.5E-8	Flood-δ_ℓ 1.7E-7	Seismic-C_1-ε 3.2E-6	
		Seismic-α 1.3E-11	Seismic-Y_1C_1-γ,δ_e 6.6E-9	Seismic-γ,δ_e 1.8E-9	Seismic-Y_1C_1-β 9.4E-10		Seismic-Y_1C_1-ε 1.2E-7	
		Seismic-C_1-α 1.3E-9		Seismic-C_1-δ_ℓ 2.3E-6				
		Seismic-Y_1C_1-α 4.7E-11		Seismic-Y_1C_1-δ_ℓ 8.4E-8				
SUM		7.37E-9	9.85E-7	1.27E-5	1.42E-7	5.25E-7	1.79E-5	7.20E-7

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Table 7.5 St. Lucie Value of Offsite Population Dose in Person-Rem for the Base Case (person-rem/reactor-year)

Release Category	Population Dose (PR) Within 50 Miles, D	Probability of Release Category, P	DP
Upper Bound Source Term (WASH 1400)			
1	1.5E+6	7.31E-9	0.0
2	1.4E+6	9.83E-7	1.4
3	1.0E+6	1.27E-5	12.7
4	6.1E+5	1.41E-7	0.1
5	3.6E+5	5.22E-7	0.2
6	7.9E+4	1.76E-5	1.4
7	6.0E+3	7.15E-7	0.0
			15.8 PR
Baseline Source Term (0.3 times WASH 1400)			
1	8.5E+5	7.31E-9	0.0
2	8.6E+5	9.83E-7	0.8
3	6.7E+5	1.27E-5	8.5
4	3.9E+5	1.41E-7	0.1
5	3.6E+5	5.22E-7	0.2
6	3.2E+4	1.76E-5	0.6
7	4.8E+3	7.15E-7	0.0
			10.1 PR
Lower Bound Source Term (0.1 times WASH 1400)			
1	5.4E+5	7.31E-9	0.0
2	6.0E+5	9.83E-7	0.6
3	4.5E+5	1.27E-5	5.7
4	2.4E+5	1.41E-7	0.0
5	1.0E+5	5.22E-7	0.1
6	1.5E+4	1.76E-5	0.3
7	4.5E+3	7.15E-7	0.0
			6.7 PR

$p(\text{SEISMIC-}Y_1C_1 - \gamma,\delta_e) = (4.69E-7)*(1.4E-2) = 6.57E-9$ RC2,

$p(\text{SEISMIC-}Y_1C_1 - \delta_e) = (4.69E-7)*(1.8E-1) = 8.44E-8$ RC3,

$p(\text{SEISMIC-}Y_1C_1 - \beta) = (4.69E-7)*(2E-3) = 9.38E-10$ RC4,

and $p(\text{SEISMIC-}Y_1C_1 - \epsilon) = (4.69E-7)*(2.5E-1) = 1.17E-7$ RC6.

Again, the example serves to demonstrate the mapping of sequences into release categories. Actually, each sequence could also be carried on to the population dose but it is easier to combine all the sequences in the release categories.

7.2.2 Alternative Estimates

The population dose estimates for the alternatives are determined by the same procedure used for the base case. The tables containing the information needed are:

	Table	Table
Alternative 1	Internal B.28	Special Emergencies 7.1
Alternative 2	Internal B.31	Special Emergencies 7.1

The results comparable to those obtained for the base case are found in Tables 7.6 and 7.7.

7.2.3 Summary of Probabilistic Risk Estimates

The results of averted offsite population dose per reactor year are summarized in Table 7.8 for the baseline source term and the upper and lower bounds.

7.3 Non-Quantifiable Values

The previous sections have outlined the value of the potential alternatives based upon probabilistic considerations. As was noted earlier, these alternatives were selected using the systematic PRA approach, however the scope of the modifications is such that they also provide value which is in many respects non-quantifiable. These potential values include such diverse areas as: the overall effect such alternatives may have on residual risk, the potential effect of some of them on equipment qualification issues and fire (Appendix R) concerns, and the possible effects on plant availability. Other issues which may well be affected are regulatory stability and decisions as to the need for changes to decay heat removal requirements. Of course, the primary objective of TAP A-45 is, in fact, to evaluate the adequacy of existing requirements. These various areas are discussed further in the following subsections.

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Table 7.6 St. Lucie Release Category Probabilities (probability per reactor-year)

	RC 1	RC 2	RC 3	RC 4	RC 5	RC 6	RC 7
Base Case	7.3E-9	9.8E-7	1.3E-5	1.4E-7	5.2E-7	1.8E-5	7.2E-7
Alternative 1	3.2E-9	4.0E-7	5.2E-6	5.9E-8	5.1E-7	7.2E-6	6.9E-7
Alternative 2	4.7E-10	6.2E-8	8.0E-7	8.9E-9	4.8E-8	1.1E-6	6.6E-8

Table 7.7 St. Lucie Expected Value of the Population Dose Within 50 Miles (person-rem per reactor year)

	Lower Bound Source Term (0.1 x WASH 1400 Source Term)	Baseline Source Term (0.3 x WASH 1400 Source Term)	Upper Bound Source Term (WASH 1400 Source Term)
Base Case	6.7	10.1	15.8
Alternative 1	2.8	4.1	6.6
Alternative 2	0.4	0.6	1.0

Table 7.8 St. Lucie Summary of Values for Averted Offsite Population Dose in Person-Rem per Reactor Year Including Internal and Special Emergencies

	Lower Bound Source Term	Baseline (Central) Source Term	Upper Bound Source Term
Alternative 1	3.9	6.0	9.2
Alternative 2	6.3	9.5	14.8

7.3.1 Reductions in Residual Risk

It was noted in Section 5.0 that the actual total core melt probability, which would include large LOCAs and ATWS, is greater than the base case numbers shown for transients and SBLOCAs. For those events which are quantified there is a threshold of probability that establishes a level below which events are "not counted". This procedure is dictated in part by the capabilities of existing analytical tools. However, many of these "uncounted" events may be affected by the support system or room (fire) vulnerabilities such as are identified here. Therefore, improvements in support systems or external event protection such as those provided in alternative 1 will (should) reduce core melt probability even more than is shown here. Furthermore, since the modifications often address potential vulnerabilities in support systems or general rooms, there also will be improvements in the response to other initiating events such as large LOCAs which have not been addressed here. Furthermore, any overall improvement in safety systems reliability will obviously reduce risk even though it may not be quantified.

It should be noted that the add-on system has a significant effect upon residual risk, whatever the source of that risk. This beneficial effect arises from the fact that the add-on is independent from other plant systems except for the required tie-ins inside containment. Therefore, the add-on is unaffected by any fires or floods which might occur elsewhere in the plant. As noted earlier, the unreliability of the add-on is a function of the availability of offsite power (OSP), and this unreliability becomes a multiplicative factor, i.e., it reduces core melt probability. The single train described here has unreliabilities of 5.2E-2 with OSP available, and 9.8E-2 without OSP. If a two train system were used in which complete independence between trains was maintained, then these unreliabilities become 2.7E-3 and 9.6E-3 respectively, providing an even greater reduction in residual risk. However, there would be an accompanying increase in costs.

7.3.2 Effects Upon Equipment Qualification

A subject of concern in many current discussions is equipment qualification, that is, whether the equipment is truly qualified to withstand the environments it may experience under accident conditions. The question arises, for example, if one considers primary system feed and bleed options for decay heat removal (see Section 10.0). It also arises if one considers piping failures that might fill a room or compartment with high temperature water or steam, or both. Similar concerns have been expressed by some with respect to inadvertant actuation of fire suppression systems that use water spray. Some of the modifications proposed in the above alternatives do deal with

0126

equipment qualification issues. For example, the modification designed to prevent fire damage to cables as the result of a cable spreading room fire insures that the cables do not see an environment for which they are not qualified. In a similar vein, the seismic-based modification which provides additional supports for the refueling water and condensate storage tanks is intended to improve system survivability in an earthquake environment. The add-on system provides an alternative approach to dealing with equipment qualification concerns. As noted above, a completely independent train of equipment can be (and is, in this design) isolated from adverse environments that might occur in the various plant buildings. Also, the separation into a dedicated building allows more stringent controls to insure that adverse environments are not created within that building. The dedicated system does not solve environmental qualification problems, if they exist, but it does reduce their impact and influence.

7.3.3 Effects Upon Plant Availability

All the modifications proposed here have improvement of decay heat removal as the goal. It is clear that some modifications could affect in-service plant availability in either a positive (improved) or negative (decreased) manner. A modification which adds equipment to a system could possibly lead to an increase in the spurious scram rate, and thus decrease availability. Or in other situations, the addition of equipment could lead to a situation in which failure of existing components with a similar function are covered by the added redundancy or diversity. Thus, in some situations, plant operation could continue without interruption, that is, technical specifications would be met while equipment was repaired rather than having to shut down while the work was being done. It seems self-evident that if the additional equipment is in the form of a standby system the probability of spurious scrams should be small. Certainly the frequency of those situations for which additional equipment could obviate a plant shutdown would be dependent upon the design of the plant and the nature of the added equipment. This area will be explored further in subsequent analyses.

7.3.4 Regulatory Issues

It has been suggested that revisions to decay heat removal capabilities could result in relaxation of requirements and perhaps contribute to increased regulatory stability. These issues should be pursued as part of the overall TAP A-45 program. However, the proper forum or agency for those evaluations is not as obvious. Certainly the technical analyses bear upon these issues, but they also require policy and regulatory perspectives which are more properly the purview of the NRC. It is also apparent that these issues should not (and perhaps cannot) be resolved on the basis of only a few

0127

plant studies. This question also will be pursued further in subsequent analyses.

7.3.5 Summary of Non-Quantifiable Values

Based upon the work to date, it is quite apparent that the modifications being proposed can reduce risks beyond that quantified here. It is equally obvious that some alternatives such as the independent add-on system provide another means to address equipment qualification issues. The effect of DHR modifications upon plant availability will be plant specific but it is being pursued. Several regulatory issues can also be affected by DHR modifications but definitive discussion requires added input in terms of plant analyses and NRC regulatory policy.

8.0 UNCERTAINTY AND SENSITIVITY CONSIDERATIONS

In this chapter the sources of uncertainty in the estimates of the "Impact" and "Value" terms discussed in Chapters 6 and 7, respectively, are examined and, where possible, quantified so that the overall uncertainty in the integrated value-impact analysis described in Chapter 9 can be evaluated.

It should be noted that in this chapter the avertable onsite costs, which are treated as "negative impacts" in Chapters 6 and 9, are treated as "positive values" as this facilitates assessment of the overall uncertainty in the value-impact analysis.

It should also be noted that there are a number of parameters for which the numerical values depend on decisions by NRC; the manner in which the value-impact ratios (see Appendix L) would vary with these parameters is indicated (e.g., whether linear or otherwise) so that the decision maker can readily examine the effect of changes in the values currently preferred by NRC.

8.1 The Nature of the Problem

It has been shown in a generic study of the treatment of uncertainty in value impact analysis[30] that a useful approach is as follows:

i) By treating avertable onsite costs as positive values instead of negative impacts, the uncertainty in the impact term is so small compared with that in the value term that it can be ignored.

ii) By analysis of the structure of the value term it can be seen which parameters can be regarded as subject to random variations, and thus can be treated statistically, and which should be regarded as empirical choices that would best be treated by means of a sensitivity analysis.

iii) If the value term is limited to offsite costs, the uncertainty can be large (see below), but if it includes onsite costs as well, the uncertainty in the results of the analysis is reduced. This is because the uncertainty in the onsite costs is smaller than that of the offsite costs and, in general, the magnitude of the onsite costs is considerably greater; consequently the large uncertainty in the offsite cost becomes relatively unimportant. The current views of the NRC Steering Committee of the Safety Goals Evaluation Program on the inclusion of onsite costs are summarized in Reference 29.

In the case of a nuclear power plant for which a PRA exists that includes an estimate of the consequences, the relative magnitudes of the offsite and onsite costs can be determined

0129

directly. Consequently, if the value term includes both onsite and offsite costs, a better appreciation of the overall uncertainty in the results of the analysis can be obtained than is possible by means of a generic approach. The availability of a plant specific PRA may also make possible a better estimate of the uncertainty in some individual parameters of the value terms, such as the probability of core melt.

As discussed in the next section, in the case of the St. Lucie NPP, the number of sequences which make a substantial contribution to the overall risk is small. This facilitates examination of the amount of the overall uncertainty attributable to the uncertainty in the probability of core melt, $p(m)_j$, due to accident sequences j, and in q_{jk}, the conditional probability of a release in category k, given core melt due to accident sequence j.

8.2 The Uncertainty in the Offsite Costs

8.2.1 The Structure of the Offsite Cost Terms

It is shown in Reference 30 that if offsite costs are evaluated in terms of \$C(D) per person rem of population dose, where C(D) is in the range \$10 to \$1000 per person rem, the offsite cost term, V(T), which is the present worth of the expectation of loss, can be represented by the following expression:

$$V(T) = n_{eff}\, C(D) \overset{\text{all } k}{\Sigma} F(D)_k \overset{\text{all } j}{\Sigma} p(r)_{jk} \qquad (1)$$

where n_{eff} is the discounted, or effective, remaining life of the plant in years.

$F(D)_k$ is a function which converts a release in category k into population dose, in averaged weather conditions, and $p(r)_{jk}$ is the probability of a release of category k due to accident sequence j. Also, $p(r)_{jk} = p(m)_j \bullet q_{jk}$ where $p(m)_j$ is the probability of core melt due to sequence j and, as noted above, q_{jk} is the conditional probability of a category k release, given core melt due to sequence j; it should be noted that the definition of j_k is NOT simply the probability of containment failure, given core melt due to sequence j. However, it may have the same numerical value.

It is also shown in Reference 30 that the change in offsite costs $\Delta V(T)$, due to an improvement which reduces $p(m)_j$ by $\Delta p(m)_j$ and q_{jk} by Δq_{jk} can be written

$$\Delta V(T) = n_{eff}\, C(D) \overset{\text{all } k}{\Sigma} F(D)_k \overset{\text{all } j}{\Sigma} [(q_{jk}\Delta p(m)_j + p(m)_j\, \Delta q_{jk})] \qquad (2)$$

The uncertainties in each of the terms on RHS of Equation (2) are discussed in the following Sections. (Note: A further problem, arising from the fact that both p(m) and q may contain unquantifiable components is discussed in Reference 31.)

8.2.2 The Uncertainty in n_{eff}

With the assumption that the remaining life of the plant, N, is its operating license length (34 years) less the current number of years of commercial operation, the only uncertainty in n_{eff} is the choice of discount rate.

In the case of St. Lucie N is 24 years and n_{eff} is

13.8 years for a discount rate of 5% and
9.0 years for a discount rate of 10%.

The choice of discount rate is at NRC's discretion, however, 5% appears to be the preferred value at the present time and this is the value used in this report, unless otherwise stated.

8.2.3 The Uncertainty in C(D)

NRC's preferred value for C(D) is $1000 per person rem of population dose, based on the total dose out to a distance of 50 miles[20]. This choice of C(D), or the distance to which population dose is integrated, might conceivably change; several studies of alternative methods for deriving the monetary value of the consequences of severe accidents are described in the literature.[33,34] The extent to which estimates based on the NRC's current guideline values for C(D) and distance differ from those derived by more complex models is discussed in Reference 30. Conceivable changes in NRC's ground rules could change the estimated monetary values by a factor of about 3 upwards or downwards. Reasonable sets of assumptions used with the more complex models could increase the estimates derived in Reference 34 by a factor of about 5, as described in Reference 31.

Clearly, the choice of C(D) and the distance to which population dose is integrated are at NRC's discretion, and consequently does not need to be considered as a source of uncertainty.

8.2.4 The Uncertainty in F(D)

In addition to the uncertainty arising from the choice of C(D) there is also some uncertainty in the conversion of a release of any given size and composition to population dose. This is discussed in Reference 30, where it is shown that application of the interdiction dose concept largely offsets most of the uncertainties arising from factors such as atmospheric dispersion, dry and wet deposition and buoyancy effects in the plume (if the population distribution is reasonably uniform) and the cost and effectiveness of decontamination processes.

0131

Consequently, the uncertainty in population dose due to these factors is small, given a reasonably uniform population distribution, although the uncertainty in property damage, due to decontamination and interdiction costs may be large.

In addition to the uncertainty in the conversion of a given category of release into population dose, there is a further uncertainty as to the correct values of the release fraction for the various isotopes occurring under the conditions which characterize that particular category. This aspect is discussed in Reference 30, where it is concluded that the fractional releases shown in Table 2.3 of Appendix K to this report may be regarded as upper bounds; for the purposes of obtaining central values for V(T) it is considered that release fractions which are one third of those shown in Table 2.3 would be a more appropriate choice and fractions which are one tenth of those in Table 2.3 could be regarded as lower bounds.

However, from the results obtained using the CRAC2 program[35] it appears that when population dose is integrated over a radius of only 50 miles, with an interdiction dose of 25 rem over 30 years, the total dose is remarkably insensitive to changes in the release fractions within the suggested upper and lower bounds. It can be seen from Table 2.4, Appendix K, of this report, that the variation about the central value for population dose is covered by a factor of 1.6 for Category 2, which is the dominant contributor to risk. Consequently, the effect of this variation on the overall uncertainty is negligible.

In the specific case of St. Lucie, the population distribution is asymmetrical, due to the oceanside location of the plant, and on the landward side the population density, out to 30 miles, in the most densely populated sector is many times the average for the other landward sectors. Thus, if a large release were to occur, the actual population dose, out to 50 miles, could be nil, or it could be substantially greater than that predicted for averaged weather conditions, depending on the wind direction at the time of the release.

Thus, if decision making can be based on averaged weather conditions, the uncertainty in F(D) is small and can be ignored. However, if it were considered necessary to take account of day to day weather variations, F(D) could vary by a factor of about 2 with respect to the central value. It is assumed in this analysis that NRC's decisions would be based primarily on averaged weather conditions.

The St. Lucie nuclear power units do not share safety systems and, therefore, there are few accident sequences for St. Lucie which could lead to core melt in both reactors. If such sequences did exist, the source terms for any given release category would be doubled. However, it can be seen from Tables

2.3 and 2.4 of Appendix K and the preceding discussion that this would have only a small effect, of order 50 percent, on population dose out to 50 miles. Thus the additional uncertainty in the offsite costs due to this possible increase in source terms would be relatively minor.

8.2.5 The Uncertainty in P(m) and q

From Equation (2), above, it can be seen that if a modification changes $p(m)_j$ without affecting q_{jk} (i.e., $\Delta q_{jk} = 0$) the equation reduces to

$$\Delta V(T) = n_{eff}\, C(D) \sum^{\text{all } k} F(D)_k \sum^{\text{all } j} q_{jk}\, \Delta p(m)_j \tag{3}$$

It is shown in Reference 17 that if the fractional improvement in q_{jk} is the same as that in $p(m)_j$, Eqn (2) becomes

$$\Delta V(T) = n_{eff}\, C(D) \sum^{\text{all } k} F(D)_k \sum^{\text{all } j} 2\, q_{jk}\, \Delta p(m)_j \tag{4}$$

The two cases, $\Delta q_{jk} = 0$ and $\Delta q_{jk}/q = \Delta p(m)_j/p(m)$ should bound the situations arising in practice; thus Equations (3) and (4) can be used to examine the effect of uncertainties in $\Delta p(m)$ and q.

From Table 7.5 of Section 7 of this report, it can be seen that most of the population dose (92%) is attributable to events giving rise to release in Categories 2 and 3 and that, of this contribution, about 16 percent is attributable to "internal" initiating events. About 62 percent of the release in Categories 2 and 3 is due to fire and another 19 percent is due to seismic events; only 3 percent is due to the other external events of flooding, wind, or lightning.

From previous PRAs it appears likely that the uncertainty in the estimated values for the probability of core melt due to internal initiating events which arises from random variations can be represented by an error factor* of about 3; the estimated probabilities for releases due to external initiating events are larger, an error factor of 10 is considered to be appropriate in this case. Taking into account the relative sizes of the contributions due to internal and external events, the estimated overall change in the probability of core melt leading to Category 2 and 3 releases is likely to be subject to an error factor of about 9. More detailed analysis of the individual sequences might lead to some refinement of this estimate.

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*Note: An error factor of 3, as used here, implies that the 95 percent confidence limits are believed to be a factor of 3 above and below the best estimate value.

0133

There is also some additional uncertainty in the upward direction due to lack of completeness in the PRA arising from unquantifiable factors such as obscure common mode faults that have not been identified. However, as we are concerned here only with differential changes in p(m) and p(r) this source of uncertainty can be ignored in the first instance. A possible method of dealing with the unquantifiable components of p(m) and q is discussed in Reference 31.

The conditional probability of releases to the environment in categories 2 and 3, given core melt can be denoted by q_2 and q_3 respectively, it is necessary to consider the uncertainties in q_2 and q_3.

A value of 1.4E-2 has been assumed for q_2 (see Table 7.2). It could conceivably be low by a factor of 2 as judged by the testimony given by NRC at the Indian Point Hearing,[36] but this seems unlikely for St. Lucie. It is considered unlikely that it is too high.

From Table 7.2 it can be seen that q_3 can be either 0.014 or 0.18 but from Table 7.4 it can be seen that for the important sequences it has the value 0.18. At the most it could therefore be too low by a factor of approximately 5; it seems unlikely that it would be too large by a factor of more than 14.

Thus the estimated values for the conditional probabilities of releases in categories 2 and 3 due to the dominant sequences are likely to be conservative. Therefore the estimated probabilities of release due to these sequences are also likely to be conservative. The effect of these uncertainties is discussed further in Section 8.3.9 below.

8.2.6 Overall Uncertainty in the Value Term V(T)

From the preceding discussion it can be seen that the main uncertainties in V(T) and their causes are as follows:

n_{eff}	Factor of about 1.5 about a best estimate value of 13.8 years, arising from a possible change in NRC's choice of discount rate in the range 0 to 10 percent.
C(D)	The NRC's empirical guidance for estimating offsite costs on the basis of $1000 per person rem of population dose is unlikely to lead to a lower estimate for St. Lucie than the more elaborate method of Aldrich and Strip[33,34] which is discussed in Reference 30. However, conceivable changes of the "ground rules" assumed by Strip could increase his estimate by a factor of about 5; e.g., changes in decontamination cost alone could lead to an

0134

	increase of about 60 percent; other factors, within the NRC's discretion, could lead to a 3-fold increase.
F(D)	If decision making can be based on averaged weather conditions, the uncertainty in F(D) would be negligible compared with that in some of the other parameters. Because of the combination of a single high population sector, and the oceanside situation of St. Lucie, there is a 50 percent chance that the population dose would be negligible, but there is a 10 percent chance that it could be at least a factor of 2 greater than the Central Estimate.
$\Sigma q_{jk} \Delta p(m)_j$	This term is likely to be subject to an error factor of about 9 in Δp_j and the further uncertainty in q_{jk} discussed in the previous Section (i.e., upper and lower bounds which are factors of 5 above and 14 below the central value of 0.2.

From the summary it can be seen that, with the existing NRC "ground rules," the dominant cause of uncertainty is the random variation in the parameters which determine $\Delta p(m)_j$ and q_{jk}. This gives rise to an error factor of about 9 in $\Delta p(m)_j$.

Possible changes in the NRC ground rules for estimating population dose could lead to an increase in V(T) by a factor of as much as 5 (choice of a zero discount rate, combined with a change in interdiction dose from 25 rem to 5 rem over 30 years). An increase in the distance is which population dose is considered could also lead to a substantial increase in the dose.

The only likely decrease in V(T) arising from a change within NRC's discretion would be that associated with an increase in the discount rate (factor of 1.5 in going from 5 percent to 10 percent per year).

Table 8.1 presents a summary of the variation of the value term with the associated uncertainties.

8.3 The Uncertainty in Combined Onsite and Offsite Costs

8.3.1 The Structure of the Onsite Cost Term

As indicated in the introduction to this Chapter, it must be noted that in the discussion of the uncertainties, "avertable onsite costs" have been treated as "positive values," not as "negative impacts."

Table 8.1

St. Lucie Nuclear Power Plant--Summary of the Variation in the "Value" Term, if Based on Offsite Costs Only, With the Principal Parameters Subject to Uncertainties

Parameter	Central Value	Likely Range	Effect on Offsite Cost	Nature of the Uncertainty
Discount Rate	5% p.a.	0% x 1.4 10% ÷ 1.5	Linear	NRC decision
Change in Core Melt Probability Δp_m	Variable	x 9 ÷	Linear	Statistical
Probability of containment failure q_i (Dominant sequences)	0.2	x 5 ÷ 14	Linear	Partly statistical, partly lack of knowledge
Change in probability of large release Δp_{ri}	Variable	x 45 ÷ 126	Linear	As q_i
Release Fractions (Dominant Sequences)	0.3 RSS	x 3 ÷	x 1.1 ÷	Lack of knowledge
Possible changes in the ground rules for estimating population dose to 50 miles	NRC Discretion	x 5 ÷ 1	Linear	NRC decision

It is shown in Reference 17 that the onsite cost term, U(T), which is the present worth of the expectation of loss, can be represented by the following expression:

$$U(T) = n_{eff}\ p(m)(X+Y+Z) \qquad (5)$$

where n_{eff} has the same meaning as in the previous Section,

p(m) is the overall probability of core melt per year

X is the differential cost of replacement power for the first 10 years after core melt

Y is the loss of capital investment remaining at the time of core melt

Z is the cost of on site cleanup after core melt

It is also shown in Reference 30 that the change in onsite cost, ΔU(T) due to an improvement which reduces p(m) by Δp(m), can be written

$$\Delta U(T) = n_{eff}\ \Delta p(m)(X+Y+Z) \qquad (6)$$

The uncertainties in each of the terms on the right hand side of Eqn (6) are discussed in the following Sections

8.3.2 Uncertainty in n_{eff}

This is the same as in the case of offsite costs.

8.3.3 Uncertainty in Δp(m)

In a single accident sequence, j, the uncertainty in $\Delta p(m)_j$ may vary widely from that of Δp(m). Detailed inspection of the individual sequences might show that the uncertainty in Δp(m) is smaller than that of the corresponding term in Equation (3) but this extension of the analysis has not yet been carried out. Thus, for the present, it is assumed that the uncertainty in the quantifiable component Δp(m) can be represented by an error factor of about 5.

8.3.4 Uncertainty in the Differential Cost of Replacement Power

As discussed in Reference 30, evaluation of the uncertainty in the differential cost of nuclear power presents difficulties of two different types: at the local level, changes in the ratio of supply to demand could have a significant effect on the cost of the replacement power from fossil fired stations (e.g., if the growth of demand were higher than expected, a higher proportion of the replacement power might have to be generated by oil fired plants or by older, less efficient, coal fired plants). At the national level escalation of either, or both, fossil or nuclear fuel costs at a rate different from the general inflation rate would lead to a change in the present worth of the replacement fuel cost, e.g., if the costs of both type of fuel were subject to an escalation rate, in real terms, of 5 percent per year, the present worth of the component of the onsite costs would be approximately twice its value in the absence of escalation.

In the case of the St. Lucie NPP, loss of one unit would represent a loss of only 16 percent of the electricity generating capacity of the coal fired units of the power pool (No. 16) which it serves[37] so that extensive use of expensive oil or gas fired units should be unlikely. Thus the uncertainty due to this cause is negligible.

In the event of an accident sequence leading to core melt in both units, the additional load demand on the coal fired units would represent 31 percent of their total capacity, leading to the use of a higher proportion of high fuel-cost plants. Thus, the cost of replacement power might increase by a factor of more than 2. However, the relative probabilities of these sequences is not high and replacement power is only 43 percent of the total onsite cost. Thus, the overall effect on the uncertainty in the onsite cost term may be small.

The escalation rates, in real terms of nuclear and fossil fuels in the No. 16 power pool, are the principal sources of uncertainty in the prediction of the replacement power costs. It seems likely that the escalation rates for both types of fuel would be at least 3 percent over the next 10 to 20 years.[38] A central value of 5 percent per year for each has therefore been used to estimate this component of the onsite costs; upper and lower limits of 7 percent and 3 percent per year should then provide reasonable upper and lower bounds for this component.

In this analysis it has been assumed that the replacement power costs in 1986 would be as estimated by VanKuiken in Reference 37. Escalation at 5 percent per year gives a present worth for the avertable loss for 10 years replacement power of $\$2.8 \times 10^{10} \bullet p(m)$ given that the accident occurs half way through the remaining plant life of 24 years. If the costs of both types of fuel were to escalate at 3 percent and 7 percent per year, the present worth would become $\$2.2 \times 10^{10} \bullet p(m)$ and $\$3.5 \times 10^{10} \bullet p(m)$, respectively, i.e., the uncertainty could be represented by a factor of 1.2 about the central value.

Other cases can be postulated in which the two fuels have different escalation rates; in the extreme cases where these are 3 percent and 7 percent, the present worth becomes $\$2.1 \times 10^{10} \bullet p(m)$ for escalation rates of 7 percent per year for nuclear and 3 percent per year for fossil fuel. In the converse case the present worth becomes $\$3.4 \times 10^{10} \bullet p(m)$.

0138

Thus, in the case where nuclear fuel costs are assumed to increase much more rapidly than fossil fuel costs, the present worth of the differential fuel costs would be only 75% of the central value. In the converse situation the central value would increase by a factor of about 1.2. For reasons which are discussed in Reference 30, it is considered unlikely that either of these situations would arise during the design lifetime of the St. Lucie NPP, particularly the case of more rapid escalation of nuclear costs.

Thus, it is considered that the overall uncertainty in the value of the replacement power cost can be represented by a factor of 1.3 about the central case in which the present nuclear and fossil fuel costs estimated in Reference 37 both escalate by 5 percent per year in real terms. It should be noted that in this central case the replacement power cost is a factor of 1.7 which would be estimated on the assumption that neither fuel cost escalated.

Inclusion in the value-impact analysis of the possibility of accidents leading to core melt in both units would not lead to a significant increase in the uncertainty in the onsite cost term.

8.3.5 Uncertainty in the Evaluation of the Loss of the Utility's Investment

In Reference 30 a method for evaluating the utility's investment at the time core melt occurs is described. The method is designed to express the value of the investment in 1985 dollars, allowing for depreciation proportional to the number of years of commercial operation preceding core melt.

Given that the method described is acceptable, the monetary value attributed to the investment should not be subject to any significant uncertainty and has not been investigated further.

8.3.6 Uncertainty in the Cost of Cleanup After Core Melt

As indicated in Section 6 of this report, the "rule of thumb" figure of $\$1.2 \times 10^9$ proposed by NRC[32] has been used in the value-impact analysis for the St. Lucie NPP. There is a general consensus of technological opinion that the cost of cleanup after core melt would be in the range $\$1 \times 10^9$ to $\$2 \times 10^9$. Thus, the "rule of thumb" figure is unlikely to be in error by a factor of more than 2.

8.3.7 Uncertainty Due to the Omission of Possible Components in the Onsite Costs

In Reference 30 a number of other possible contributions to the onsite cost term are discussed. Inclusion of these items could increase the total onsite costs due to the 3 components discussed above (i.e., the terms X, Y, and Z in Equation 5) by a

0139

factor of about 2. The decision as to whether or not these additional costs should be included rests with NRC.

8.3.8 Summary of the Uncertainty in the Onsite Cost Term

By summing the lower and the upper bound values for the 3 onsite monetary terms (X, Y, and Z of Equation 5) overall limits to the monetary terms are found to be $\$3.95 \times 10^{10}$ and $\$6.6 \times 10^{10}$. Thus a reasonable estimate for the central value is $\$4.8 \times 10^{10}$, with upper and lower bounds a factor 1.3 above and below the central value.

The uncertainties in the various components of the onsite cost terms are summarized in Table 8.2. It can be seen that the overall uncertainty is dominated by that of Δpm; the next most important contribution arises from items where NRC decisions are required; the uncertainty in the monetary terms (X, Y, and Z) is relatively unimportant.

8.3.9 The Uncertainty in the Combined Onsite and Offsite Costs

8.3.9.1 The Relative Sizes of the Offsite and Onsite Costs

In other chapters of this report central values are given for the present worths of the maximum expectations of loss attributable to avertable offsite and onsite costs associated with possible failures of DHR related systems of the St. Lucie NPP in its present form. These central values are as follows:

Offsite costs	$\$1.39 \times 10^5$
Onsite costs	$\$2.67 \times 10^6$

Thus it can be seen that the onsite costs are about 15 times as large as the offsite costs. It is to be expected that the ratio of the two types of costs would be similar after each of the various sets of modifications which have been examined.

In accident sequences that could lead to core melt in both units the onsite costs would be doubled but, as noted in Section 8.2.4, above, population dose would be unlikely to increase by more than 10 percent. Thus the ratio of offsite to onsite costs would be reduced by a factor of about 1.8.

8.3.9.2 The Uncertainty in the Combined Offsite and Onsite Costs

Since the overall uncertainty in the onsite terms is substantially less than that in the offsite terms and the size of the former is about 15 greater, the overall uncertainty in the total averted costs associated with a particular set of modifications will be nearer to that of the onsite term. Using the results obtained in other Sections, this can be demonstrated quantitatively for 2 of the Alternatives (the other one yields

Table 8.2

St. Lucie Nuclear Power Plant--Summary of the Variation in the "Value" Term, if Based on Onsite Costs Only, With the Principal Parameters Subject to Uncertainties

Parameter	Central Value	Likely Range	Effect on Onsite Cost	Nature of the Uncertainty
Discount Rate	5% p.a.	0% x 1.4 10% ÷ 1.5	Linear	NRC decision
Change in Core Melt Probability Δp_m	Variable	x 9 ÷	Linear	Statistical
Change in the monetary terms in the Value analysis	See Value analysis	x 1.3 ÷	Linear	Lack of knowledge
Possible additional contributions to onsite costs	See Value analysis	x 2.0 ÷ 1.0	Linear	NRC decisions

similar results), as shown in Table 8.3. It should be noted that Table 8.3 does NOT include the effects of possible changes which are at the discretion of NRC.

In completing Table 8.3 the upper and lower bounds for offsite and onsite costs have been estimated using the uncertainty factor in Tables 8.1 and 8.2 for each parameter (other than those where a decision by NRC is concerned) combined together, i.e., the factors (x and +) shown in the third columns of Tables 8.1 and 8.2, as representing the "Likely Range" in the various parameters for which probability distributions are not known, have been applied to the estimated 95 percent confidence limits of Δp_m to obtain bounding values for $\Delta u(T)$ and $\Delta V(T)$.

The sensitivity of the results to changes in the assumptions made about NRC's decisions concerning certain parameters can be seen by examining their effects on the central values in Table 8.3, viz:

i) Increasing the averted offsite costs by a factor of 5, (e.g., to take account of possible changes in the NRC ground miles for estimating population dose out to 50 miles) would only increase the combined averted cost by a factor of 1.2.

ii) Increasing the averted onsite costs by a factor of 2, to take account of other possible causes of loss, would increase the combined averted cost by about 95%.

iii) A change in discount rate from 5 percent to 10 percent per year would reduce all averted costs by a factor of 1.5.

iv) A change to zero discount rate for averted offsite costs only would increase these by a factor of 1.6 but would only increase the combined averted costs by a factor of about 1.03.

A change in method used to estimate offsite and onsite avertable costs, from NRC's empirical rules to the less empirical method used by Strip (i.e., basically the same as in the CRAC2 code, for offsite costs), would not have a major effect on the overall uncertainty.

8.4 Effects of Uncertainties in the Impact Term

The principal sources of uncertainty in the impact term are:

i) Uncertainties in the costs of the various "Alternatives."

ii) The effects of the various "Alternatives" on the availability of the NPP during the remainder of its service life.

0142

Table 8.3 St. Lucie Nuclear Power Plant

Uncertainty in Combined Offsite and Onsite Costs (Present Worths in Dollars) due to Factors NOT Dependent on NRC Decisions

Alternative No.	Averted Offsite Cost			Averted Onsite Cost			Combined Averted Cost			Uncertainty Factor in Combined On- and Offsite Costs
	Central Value	Lower Bound	Upper Bound	Central Value	Lower Bound	Upper Bound	Central Value	Lower Bound	Upper Bound	
		÷ 126	x 45		÷ 11.7	x 11.7				
1	$0.0828x10^{6}$	$6.57x10^{2}$	$3.73x10^{6}$	$1.49x10^{6}$	$1.27x10^{5}$	$1.74Ex10^{7}$	$1.57x10^{6}$	$1.28x10^{5}$	$2.11x10^{7}$	x 13.4 ÷ 12.2
3	$0.131x10^{6}$	$1.04x10^{3}$	$5.89x10^{6}$	$2.51x10^{6}$	$2.14x10^{5}$	$2.94x10^{7}$	$2.64x10^{6}$	$2.15x10^{5}$	$3.53x10^{7}$	x 13.3 ÷ 12.2

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As discussed in Reference 30, the uncertainty in the cost estimates for the "Alternatives" depends largely on the degree of detail to which the design of the alternative is carried.

It is not expected that any of the "Alternatives" except Number 2 would have any significant effect on availability. As alternative 2 provides another, independent, system for removing decay heat, it is possible that it could improve availability by avoiding the need for a shutdown, in order to comply with the Technical Specifications for the NPP, following the loss of one of the other systems for decay heat removal.

In the case of St. Lucie a saving of only 1 day's outage per unit per year, on average, over the remaining life of the station, would have a present worth of $\$5.5 \times 10^6$. The present worths of other savings would be pro rata.

In principle, it should be possible to evaluate the number of days that could be saved at St. Lucie with relatively little error, since plant specific failure rates should be obtainable. However this has not yet been attempted. Consequently it is not possible at this stage to discuss the effect of this part of the impact term on the overall uncertainty in the analysis.

8.5 The Overall Uncertainty in the Value-Impact Analysis

For the reasons discussed in Reference 30, when the uncertainty in either the value terms or the impact terms (or both) is large, it is more useful to express the results of the analysis in terms of a non-dimensional ratio rather than as a "Net Benefit" in dollars. In this case we have an uncertainty in the Value term which can be represented by upper and lower bounds that are approximately a factor of 30 on either side of the central value whereas in the Impact term the corresponding bounds would be a factor of less than 2 about the central value. Whatever the forms of the frequency distribution for the Value and Impact terms, it seems highly likely the upper and lower bounds on their ratio would be at a factor of about 3 on either side of the mean. For example, for illustrative purposes, if it were assumed that each term could be represented by log normal functions with Error Factors of 3 and 2 respectively, the Error Factor of their ratio would be 32.

9.0 INTEGRATED VALUE-IMPACT ANALYSIS

In this section the quantitative value analysis (Section 7.0) and the impact analysis (Section 6.0) are combined to form an integrated value-impact analysis. Unquantifiables are not treated in this section but are addressed in various other sections. The goal of this assessment is to provide measures that can be used by the NRC to make decisions on the adequacy of decay heat removal requirements (USI A-45). The methodology for the value-impact analysis is presented in Appendix L. In order to implement the value-impact analysis and illustrate the steps performed, the methodology will be reviewed briefly prior to summarizing the St. Lucie results.

9.1 Methodology

9.1.1 Value and Impact Analysis Variables

Each of the variables to be used as input in the V-I analysis is defined in Table 9.1 and characterized in several ways. First, the costs and values may be incurred one time or on an annual basis. All recurring costs or costs which might occur at any time during the remaining plant lifetime must be present valued. The present worth factor for St. Lucie based upon 24 years remaining life is 13.8 at a 5% discount rate. Second, there are positive and negative values and impacts. For example, in the case of I_4 and V_4 the modifications could result in an increased or decreased value or impact. The modifications could require additional maintenance resulting in increased radiation exposure and/or longer outage periods and thus more replacement power costs. On the other hand, the modifications could make maintenance easier, thus reducing radiation exposure and/or replacement power costs. Third, costs or doses result from either the proposed modifications or an accident. Fourth, each variable may affect the utility, and the NRC, or the public. Last, the information for each variable comes from the architect engineer, the value analysis, or other, previously documented, analyses.

In addition to the value and impact variables, the change (reduction) in core melt probability (Δp_m) for each alternative from the base case core melt probability (i.e., without any modifications) is used.

$$\Delta p_m(j) = p_m - p_m(j), \text{ central value}$$

where p_m = base case core melt probability, and

$p_m(j)$ = core melt probability of the jth alternative.

These values are given in Section 7.1.3. The upper (U) or lower (L) bounds for the change in core melt probability are determined as follows:

0145

Table 9.1 Value and Impact Analysis Input Variables

Symbol	Description		Results from	Affects the	Source of Information
I_1	Engineering and Installation Cost one time	Positive Impact	Modifications	Utility	Architect Engineer
I_2	Operations and Maintenance Costs/year - present worth	Positive Impact	Modifications	Utility	Architect Engineer
I_3	Installation Replacement Power Costs - one time	Positive Impact	Modifications	Public	Architect Engineer
I_4	In-service Replacement Power Costs - per year - present worth	Positive or Negative Impact	Modifications	Public	Not available - but probably negligible
I_5	Avertable Onsite Costs - one time - present worth	Negative Impact	Accident	Utility	Based on previous analyses
I_{51}	Replacement Power Costs				
I_{52}	Loss of Investment Costs				
I_{53}	Site Cleanup Costs				
I_6	Other Costs - one time	Positive Impact	Modifications	Utility & NRC	Not covered in this program
V_1	Averted Onsite Dose Over Plant Lifetime	Positive Value	Accident	Utility	Based on previous analyses
V_1'	Present Worth of Averted Onsite Dose @ $1000/p-rem	Positive Value	Accident	Utility	Based on previous Analyses

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Table 9.1 Value and Impact Analysis Input Variables (continued)

Symbol	Description		Results from	Affects the	Source of Information
V_2	Averted Offsite Dose Over Plant Lifetime	Positive Value	Accident	Public	From value analyses
V_2'	Present Worth of Averted Offsite Dose @ $1000/p-rem	Positive Value	Accident	Public	From value analyses
V_3	Installation Dose - one time	Negative Value	Modifications	Utility	Architect Engineer
V_3'	Present Worth of Installation Dose @ $1000/p-rem	Positive Value	Modifications	Utility	Architect Engineer
V_4	In-service Operational Dose over Plant Lifetime	Positive or Negative Value	Modifications	Utility	Not available
V_4'	Present Worth of Occupational Dose @ $1000/p-rem	Positive or Negative Value	Modifications	Utility	Not available

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$$\text{Upper bound } \Delta p_m(j) = \Delta p_m(j) \times 8$$

$$\text{Lower bound } \Delta p_m(j) = \Delta p_m(j) \div 8$$

The factor of 8 was estimated by using an error factor of 3 for the internal event sequences and an error factor of 10 for the special emergency sequences as shown in Table 9.2. When these error factors are applied to the central value or best estimate for each initiator, i.e., internal, seismic, flood, etc., and summed, the overall core melt bounds for the initiators considered in this program are 1.07E-5 for the lower bound and 6.45E-4 for the upper bound. While these bounds are not exactly symmetrical about the central value, the range is approximately 60 resulting in an error factor of 8.

9.1.2 Value Impact Analysis Measures

Impact Measures - Impacts I_2 and I_4 (refer to Table 9.1) must be multiplied by the present worth factor, 13.8 for St. Lucie assuming 5% discount rate. The present worth factor accounts for the reduced worth of the payments made or incurred at some future date. Therefore:

Total Positive Impact of the alternative, $TI(j) = I_1(j) + 13.8\ I_2(j) + I_3(j) + 13.8\ I_4(j)$.

This is the total of all the positive impacts. The negative impacts sum to the,

Total avertable cost, $I'_5(j) = I'_{51}(j)\ I'_{52}(j) + I'_{53}(j)$.

where

$$I'_{51}(j) = \Delta p_m(j) \times I_{51}(j)$$

$$I'_{52}(j) = \Delta p_m(j) \times I_{52}(j)$$

$$I'_{53}(j) = \Delta p_m(j) \times I_{53}(j)$$

Upper and lower bounds are introduced at this point in the impact analysis summary using the error factor of 8 discussed previously. That is,

Lower bound (L) $I'_5(j) = I'_5(j) \div 8 = I_5(j) \times \Delta p_m(j) \div 8$,

Central value (C) $I'_5(j) = I_5(j) \times \Delta p_m(j)$, and

Upper bound (U) $I'_5(j) = I'_5(j) \times 8 = I_5(j) \times \Delta p_m(j) \times 8$.

The same is true for $I'_{51}(j)$, $I'_{52}(j)$, and $I'_{53}(j)$. Therefore, for the jth alternative,

Table 9.2 St. Lucie Value-Impact Analysis Error Factor

	Core Melt Probability		
Analysis	**Lower Bound**	**Central**	**Upper Bound**
Internal	4.7E-6	1.4E-5	4.2E-5
Fire	4.4E-6	4.4E-5	4.4E-4
Internal Flood	ε	ε	ε
Wind & Missiles	ε	ε	ε
External Flood	3.2E-7	3.2E-6	3.2E-5
Seismic	1.3E-6	1.3E-5	1.3E-4
Lightning	2.0E-8	2.0E-7	2.0E-6
	1.07E-5	7.44E-5	6.46E-4

0149

Net Impact = Total Impact - Total Avertable Impacts or

$NI(j) = TI(j) - I_5'(j)$.

The discounted values for the negative impacts due to avertable onsite costs, conditional upon an accident, from Appendix L are:

I_{51} = \$1.56E+10,

I_{52} = \$3.7E+9, and

I_{53} = \$1.66E+10 in 1985/1986 dollars.

Value Measures - The averted onsite dose (V_1) for each alternative is estimated from onsite dose received during an accident. For purposes of this analysis, this onsite dose (51,000 person-rem) is assumed to be the same for any core melt accident as discussed in Appendix L. The onsite dose (51,000 person-rem) is multiplied by the Δp_m for the jth alternative and by the number of years of operation remaining. Thus,

$V_1(j) = (51000) \times \Delta p_m(j) \times 24$.

The present worth, $V_1'(j)$, of the above avertable onsite dose, valued at \$1000 per person-rem is:

$V_1'(j) = (51000) \times \$1000 \times \Delta p_m(j) \times 13.8$.

The offsite averted dose (V_2) for each alternative is estimated from the avertable offsite dose per reactor year (given for each alternative in Section 7.0) multiplied by the remaining years of plant operation. Thus

$V_2(j)$ = (Averted doses from Table 7.8) x 24 .

In this case the Δp_m is not required because the core melt probability is inherent in the calculations in Section 7.

The present worth $V_2'(j)$ of the avertable offsite dose valued at \$1000 per person-rem is:

$V_2'(j)$ = (Averted doses from Table 7.8) x \$1000 x 13.8.

The totals of the positive values (onsite + offsite) for averted dose and costs are:

$V_{12}(j) = V_1(j) + V_2(j)$

$V_{12}'(j) = V_1'(j) + V_2'(j)$.

The ratio of the averted offsite dose (V_2) from Table 7.8 and the base case dose from Table 7.7 is

Averted dose ratio, offsite ≡

$$ADR_o = \frac{V_2(ji)}{\text{Base Case Dose (i)} \times 24}$$

where i represents the lower, central, and upper bound values.

Similarly, the ratio of the total averted dose (V_{12}) and the total base case dose is

Averted dose ratio, onsite and offsite ≡

$$ADR_n = \frac{V_{12}(ji)}{\text{Base Case Dose (i)} \times 24 + V_1(j)}$$

The negative values considered for the jth alternative included the dose received during installation (a one time dose), $V_3(j)$, and the in-service occupational dose received over the remaining plant life time, $V_4(j)$. These doses can be presented valued at $1000 per person-rem in a manner analogous to that used for the positive values, thus,

$$V_3'(j) = V_3(j) \times \$1000$$

$$V_4'(j) = (V_4(j)/24) \times \$1000 \times 13.8 \quad .$$

$V_3'(j)$ does not include the present worth factor (13.8) because it is a one time dose, whereas the dose associated with operations, $V_4(j)$ is recurring.

The net averted dose, $NV(j)$, and net averted costs, $NV'(j)$ associated with each alternative can be calculated by subtracting the negative values from the positive. Thus

$$\text{Net averted dose} = NV(j) = V_1(j) + V_2(j) - V_3(j) - V_4(j)$$

$$\text{Net present worth} = NV'(j) = V_1'(j) + V_2(j) - V_3'(j) - V_4(j)$$
of averted costs
at $1000/p-rem

Value-Impact Measures - The value-impact measures can be constructed from the variables defined above. Each of the value-impact measures is calculated with the central cost for the total impact (TI) and the central cost for the net impact (NI) as opposed to using the upper and lower bounds for NI.

The first measure considered is the ratio of averted costs to impacts. This value-impact ratio (VIR) is calculated twice.

One ratio considers only the averted offsite costs and the total impact. Thus for the jth alternative

$$VIR_o \text{ (offsite)} = V_2'(j)/TI(j)$$

The second ratio is the net value-impact ratio which accounted for the net averted costs and the net impacts.

$$VIR_n \text{ (offsite and onsite)} = NV'(j)/NI(j)$$

Similarly there are two net benefit values. The first again considers only averted offsite costs, while the second uses net averted costs and impacts. So one has,

$$NBV_o \text{ (offsite)} = V_2'(j) - TI(j)$$

$$NBV_n \text{ (offsite and onsite)} = NV'(j) - NI(j)$$

The final value-impact measure presented is the estimated cost in dollars per person-rem of Dose Averted if the alternative is implemented. Again there are two values, one considering only the offsite averted costs, the second the net costs.

$$DPR_o \text{ (offsite)} = TI(j)/V_2(j)$$

$$DPR_n = \text{(offsite and onsite)} = NI(j)/NV(j)$$

A more complete discussion of the reasons for selecting these measures is provided in Appendix L.

9.2 Results

The values, impacts, and value-impact measures defined in the previous section were calculated for the St. Lucie plant using the results of the internal analysis, special emergency analyses, and the impact analysis. These results are tabulated in Tables 9.3, 9.4, and 9.5. It is important to note that the offsite population dose is an integrated dose out to a radius of 50 miles from the site and the conversion from dose to cost is at $1000 per person rem. All present value estimates are based upon a 5% discount rate. Each of these tables gives the upper (U), central (C), and lower (L) bounds when applicable.

The symbols for the values, impacts, and measures are given at the heading of each column to avoid any possible confusion about the entries in the table or how the numbers were determined.

Table 9.3 summarizes the impacts for St. Lucie by alternative. The four positive impacts are individually tabulated and then totaled to obtain the Total Impact (TI) of modifications associated with each alternative. One time costs of installing the modifications and replacement power during the installation are already in present worth dollars. The in service operations

TABLE 9.3 ST. LUCIE - SUMMARY OF IMPACTS (Based Upon 5% Discount)

		POSITIVE IMPACTS ASSOCIATED WITH MODIFICATIONS (Present Worths)					NEGATIVE IMPACTS DUE TO AVERTABLE ONSITE COSTS (Present Worths)				
		Utility Costs		Replacement Power Costs							
Alternative No.	Change in Core Melt Probability [Central Value]	Installation and Engineering Costs ($ $\times 10^{-6}$)	Operations and Maintenance Costs (PW) ($ $\times 10^{-6}$)	Installation ($ $\times 10^{-6}$)	In Service (PW) ($ $\times 10^{-6}$)	TOTAL IMPACT ($ $\times 10^{-6}$)	Replacement Power Costs ($ $\times 10^{-6}$)	Loss of Investment Costs ($ $\times 10^{-6}$)	Site Cleanup Costs ($ $\times 10^{-6}$)	Total Avertable Costs ($ $\times 10^{-6}$)	NET IMPACT ($ $\times 10^{-6}$)
							[U, C, & L Values]				
	Δp_m	I_1	$13.8\ I_2$	I_3	$13.8\ I_4$	TI	I'_{51}	I'_{52}	I'_{53}	I'_5	NI
1	4.14E-5	.881	.207	0.0	Not Available Probably Negligible	1.088	L 0.08	0.02	0.09	0.19	0.898
							C 0.65	0.15	0.69	1.49	- 0.402
							U 5.17	1.23	5.50	11.90	-10.812
2	6.99E-5	53.791	4.802	0.0	"	58.593	L 0.14	0.03	0.15	0.32	58.273
							C 1.09	0.26	1.16	2.51	56.083
							U 8.72	2.07	9.28	20.07	38.523

$TI = I_1 + 13.8\ I_2 + I_3 + 13.8\ I_4$

$I_5' = I_{51}' + I_{52}' + I_{53}'$

$NI = TI - I_5'$

Note: L, C, and U represent the lower, central, and upper bound estimate for core melt probability

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TABLE 9.4-a ST. LUCIE - SUMMARY OF VALUES (Based on Population Dose to 50 Miles, 5% Discount Rate)

Alternative No.	Change in Core Melt Probability [Central Value]	POSITIVE VALUES Onsite Averted Dose (p-rem)	Onsite Present Worth of Averted Dose @ \$1000/p-rem (\$ x10^{-6})	Offsite Averted Dose (p-rem)	Offsite Averted Dose + Base Case Dose	Offsite Present Worth of Averted Dose @ \$1000/p-rem (\$ x10^{-6})	Total Averted Dose (p-rem)	Total Averted Dose + Base Case Dose	Total Present Worth of Averted Dose @ \$1000/p-rem (\$ x10^{-6})
				[U, C, & L Values]			[U, C, & L Values]		
	Δp_m	V_1	V_1'	V_2	ADR_o	V_2'	V_{12}	ADR_n	V_{12}'
1	4.14E-5	51	0.029	L 94	.58	.0538	145	.68	.083
				C 144	.59	.0828	195	.66	.112
				U 221	.58	.1270	272	.63	.156
2	6.99E-5	86	0.049	L 151	.94	.0869	237	.96	.136
				C 228	.94	.1311	314	.96	.180
				U 355	.94	.2042	441	.95	.253

V_1 = 51000 x Δp_m x 24
V_1' = 51000 x Δp_m x \$1000 x 13.8
V_2 = Averted Dose x 24
V_2' = Averted Dose x \$1000 x 13.8
ADR_o = V_2 + (Basecase Dose x 24)
V_{12} = $V_1 + V_2$
ADR_n = V_{12} + (Basecase Dose x 24 + V_1)
V_{12}' = $V_1' + V_2'$

Note: L, C, and U stand for lower, central, and upper bound source term estimate

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TABLE 9.4-b ST. LUCIE - SUMMARY OF VALUES (Based on Population Dose to 50 Miles, 5% Discount Rate)

		NEGATIVE VALUES						Net Value	
		Installation		Operation		Total			
Alternative No.	Change in Core Melt Probability [Central Value]	Installation Dose (p-rem)	Present Worth of Installation Dose @ $1000/p-rem	In-Service Operational Dose (p-rem)	Present Worth of In-Service Oper. Dose ($ x 10^{-6})	Installation and Operational Dose	Present Worth of Install. & Oper. Dose @ $1000/p-rem ($ x10^{-6})	Averted Dose (p-rem)	Present Worth of Averted Dose @ $1000/p-rem ($ x10^{-6})
								[U, C, & L Values]	
	Δp_m	V_3	V_3'	V_4	V_4'	$V_3 + V_4$	$V_3' + V_4'$	NV	NV'
1	4.14E-5	0	.000	Negligible		0	.000	L 145 C 195 U 270	.083 .112 .156
2	6.99E-5	20	.020	Negligible		20	.020	L 217 C 294 U 421	.116 .160 .233

V_3' = V_3 x \$1000

V_4' = $(V_4 + 24)$ x \$1000 x 13.8

NV = $V_1 + V_2 - V_3 - V_4$

NV' = $V_1' + V_2' - V_3' - V_4'$

Note: L, C, and U stand for lower, central, and upper bound source term estimate

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TABLE 9.5 ST. LUCIE - SUMMARY OF VALUE-IMPACT ANALYSIS (Based on Population Dose to 50 Miles; 5% Discount Rate)

				V-I ANALYSIS BASED ON OFFSITE COSTS					V-I ANALYSIS BASED ON OFFSITE AND ONSITE COSTS				
						Measures of V-I					Measures of V-I		
Alternative No.	Change in Core Melt Probability [Central Value]	Offsite Averted Dose (p-rem) [U,C,&L Values]	Total Averted Dose + Base Case Dose [U,C,&L Values]	Total Impact [Central Cost] ($ x10^{-6})	Present Worth of Averted Dose @$ 1000/ p-rem [U,C,&L Values] ($ x 10^{-6})	V-I Ratio	Net Benefit ($ x 10^{-6})	Dollars per p-rem	Net Impact [Central Cost] ($ x10^{-6})	Present Worth of Averted Dose @ $1000/ p-rem [U,C,&L Values] ($ x10^{-6})	V-I Ratio	Net Benefit ($ x10^{-6})	Dollars per p-rem
	Δp_m	V_2	ADR_n	TI	V_2'	VIR_o	NBV_o	DPR_o	NI	NV'	VIR_n	NBV_n	DPR_n
1	4.14E-5	L 94	.68	1.088	.0538	.049	- 1.034	11574	- 0.402	.083	- .206	.485	0
		C 144	.66		.0828	.076	- 1.005	7556		.112	- .279	.514	0
		U 221	.63		.1270	.117	- .961	4923		.156	- .388	.558	0
2	6.99E-5	L 151	.96	58.593	.0869	.001	-58.506	388033	56.083	.136	.002	-55.947	258447
		C 228	.96		.1311	.002	-58.462	256987		.180	.003	-55.903	190759
		U 355	.95		.2042	.003	-58.389	165051		.253	.004	-55.830	133214

$VIR_o = V_2' + TI$
$NBV_o = V_2' - TI$
$DPR_o = TI + V_2$

$VIR_n = NV' + NI$
$NVB_n = NV' - NI$
$DPR_n = NI + NV$

Note: L, C, and U stand for lower, central, and upper bound source term estimate

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and maintenance costs and replacement power costs must, however, be multiplied by 13.8 to account for the present worth of these impacts. The installation of modifications at St. Lucie can be accomplished during normal outages so that there are no replacement power costs. Although replacement power costs due to in service maintenance were not specifically estimated, our judgment is that these costs will have a negligible impact. It is obvious from the table that alternative 2 will cost substantially more than alternative 1 due to the extent of the modifications.

The negative impacts result from the averted onsite costs attributed to a potential accident. The costs are, of course, probabilistic. Thus the potential costs for I_{51}, I_{52}, and I_{53} in 1985/1986 dollars must all be multiplied by the lower (L), central (C), and upper (U) bound values for Δp_m. The net impact (NI) is the positive impacts minus the negative impacts. The lower the net impact the more favorable the alternative appears. In fact, in a few cases the net impact is negative which means that the averted onsite costs due to a possible accident are greater than the cost of installing the modifications for those alternatives.

Table 9.4a presents the positive values for each alternative. These are onsite averted dose (V_1) and offsite averted dose (V_2) due to a potential accident. These are both probabilistic in nature, however, the calculation of offsite averted dose (V_2) does not explicitly include Δp_m since it is implicitly included in the analysis to obtain V_2. Upper and lower bounds are given for V_2 as derived from the risk analysis in Section 7 and based on the source term bounds assumed in Section 7. Both averted doses must include a factor to account for the remaining plant life which is 24 years in 1986.

The total averted dose (V_{12}) is also given as are the present worth dollar values. Each of the dollar values is based on \$1000/p-rem and a 13.8 percent worth factor as described in Appendix L.

The averted doses increase from alternative 1 to alternative 2 due to the increase in Δp_m from alternative 1 to alternative 2 but there is little difference between the offsite and total averted dose ADRs.

Table 9.4b summarizes the negative values for each alternative. The installation dose (V_3) results from radiation exposure to contractor personnel during installation of the modifications for any particular alternative. The V_3 value for alternative 2 is slightly higher since this alternative requires entry into containment whereas alternative 1 does not. In-service occupational dose is considered to be negligible at St. Lucie for the alternatives proposed here. In each case the doses are

0157

converted to dollars by \$1000/p-rem. V_3 is already a present worth but V_4 requires a 13.8 present worth factor.

The net value is the positive values V_{12} or V_{12}' minus the negative values for $V_3 + V_4$ and $V_3' + V_4'$ respectively. Upper and lower bounds are given for NV and NV' which result from the bounds from the positive values only.

Table 9.5 is a summary of the Value-Impact analysis and as such several measures are repeated from Tables 9.3, 9.4a, and 9.4b. These repeated measures are TI, NI, V_2, V_2', ADR_n, and NV'. The value-impact measures derived from these measures are the Value-Impact Ratio (VIR), the Net Benefit Value (NBV), and the Dollars per person-rem (DPR) based on offsite costs alone (subscript o) and based on offsite and onsite costs combined (subscript n). Upper and lower bounds are shown where applicable. Recall that the V_2, V_2', ADR_n, and NV' value measure bounds are derived from the source term bounds discussed in Section 7 (that is approximately a ± 3 error factor). The impact measures TI and NI are central values even though upper and lower bounds are shown for NI in Table 9.3 to illustrate the range these onsite costs could have.

Table 9.6 is a further condensation of the results which show eight measures extracted from Table 9.5 for the central value only. This part of the analysis does not consider any of the non-quantifiable values. A discussion of the non-quantifiable values is given in Section 10.

0158

TABLE 9.6 ST. LUCIE - SUMMARY OF VALUE-IMPACT MEASURES (Central Value)

Alter-native No.	Change in Core Melt Probability	Offsite Averted Dose (p-rem)	Total Averted Dose Ratio	V-I Analysis Based on Offsite Costs			V-I Analysis Based on Offsite and Onsite Costs		
				V-I Ratio	Net Benefit ($ x10^{-6})	Dollars per p-rem	V-I Ratio	Net Benefit ($ x10^{-6})	Dollars per p-rem
	Δp_m	V_2	ADR_n	VIR_o	NBV_o	DPR_o	VIR_n	NBV_n	DPR_n
1	4.14E-5	144	.66	.076	- 1.005	7556	- .279	.514	0
2	6.99E-5	228	.96	.002	-58.462	256987	.003	-55.903	190759

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10.0 SPECIAL ISSUES RELATED TO DECAY HEAT REMOVAL

There has been considerable interest for many years in the use of feed and bleed (F&B) on the primary system as an alternative, essentially last resort, measure to remove decay heat from the reactor core. In addition, during the course of this investigation and other studies related to steam generator tube ruptures, questions have arisen related to the use of rapid secondary side blowdown to remove decay heat and depressurize the primary sufficiently to permit residual heat removal system cut-in. These two issues are discussed here, including some results from thermal-hydraulic studies on both subjects conducted on behalf of TAP A-45 by the Los Alamos National Laboratory.

10.1 Feed and Bleed

Feed and bleed is used more or less generically to describe two similar though somewhat different processes. In some plants, primarily those designed by Babcock and Wilcox, the high pressure injection (HPI) pumps have sufficient head and capacity to lift primary safety relief valves (SRVs) in addition to the power operated relief valves (PORVs) and provide sufficient flow to remove decay heat. Therefore, they can literally feed and bleed with liquid only (no void in the primary). In other plants the HPI pumps do not have sufficient head to lift the SRVs and so the PORVs must be opened first to reduce primary pressure to the point where HPI can provide adequate flow. In these plants the feed is liquid, but the bleed is usually steam or a two phase mixture, i.e., there will be some void in the primary. Detailed thermal-hydraulic analysis of feed and bleed phenomena and capabilities are reported elsewhere.[13,14,15,16] In general, those studies support the concept of feed and bleed but they do point out that timing is a critical parameter in establishing whether or not primary feed and bleed can successfully remove decay heat.

10.1.1 Value of Feed and Bleed

In the case of St. Lucie Nuclear Plant, as a result of discussions with the plant staff which indicated that there were plant specific procedures for bleed and feed in place in January 1986, the base case analysis was established with bleed and feed. Therefore, in constructing the event trees (Appendix B) credit was given for this capability. As reported in Appendix B, the ability to feed and bleed can influence thirty-one potential accident sequences. If the plant cannot feed and bleed these thirty-one sequences reduce to twelve sequences. The specifics of the sequences replaced are discussed in Appendix B. It is sufficient for this discussion to note that the combined internal core melt probabilities considering only the affected sequences after recovery analysis are:

0160

P(cm) (affected sequences without feed and bleed)	4.26E-5
P(cm) (affected sequences with feed and bleed)	8.73E-6

Therefore, a feed and bleed capability reduces the estimated core melt probability for internal events at St. Lucie by 3.39E-5 per reactor year. It is noted that this result is substantially greater than several other recent studies. For example, Sandia estimated a reduction in core melt probability of 4.0E-6 when examining the cost-benefit of adding PORVs to System 80 plants.[17] However, in that study common mode failures in pumps and valves were not modelled as extensively as they are in the present work.

There are aspects of the feed and bleed which merit additional consideration for several reasons. First, the inclusion of common mode considerations in this revised analysis has increased the value of feed and bleed significantly with respect to the prevention of core melt. Second, inclusion of common mode considerations has led to different failures dominating the system cut sets. Third, the apparent value of feed and bleed must be examined in light of phenomenological considerations arising from recent thermal-hydraulic studies.

If the St. Lucie analysis is performed with the mechanical failures (e.g., pumps, valves) not including common mode, (that is, essentially the only common mode failures considered are those for electrical power systems) electrical power failures are the dominant causes of system failures. For example, in that analysis for the single sequence, $T_1M\bar{P}QL$, it is found that if feed and bleed is viable there are three possible sequences; $T_1M\bar{P}Q\ L\bar{D}_1\bar{P}_1H_1$, $T_1M\bar{P}Q\ L\bar{D}_1P_1$, $T_1M\bar{P}Q\ LD_1$. In the sequence $T_1M\bar{P}QL_1$ without feed and bleed, the dominant contribution to core melt comes from electric power failures. When the individual cut sets are examined it is found that the majority of the estimated core melt probability is attributable to battery failures, a battery failure in combination with a component failure, or battery failures in combination with diesel generator failures. When attention is turned to the three feed and bleed sequences, it is found that the first one, $T_1M\bar{P}QL\bar{D}_1\bar{P}_1H_1$, contributes a negligible amount to estimated core melt probability. This is due to the fact that it is, by definition, a path on the event tree which has significant system successes. Even though auxiliary feedwater (L) has failed, high pressure injection (D_1) and the power operated relief valves (P_1) succeed and only the late time failure of high pressure recirculation (H_1) becomes a factor. It should also be noted that success of D_1 and P_1 also implies that in this sequence certain electrical failures cannot occur in L. The second sequence, $T_1M\bar{P}QL\bar{D}_1P_1$, contributes about one quarter of the three sequence total core melt probability. Again, for this sequence the failures are combinations of auxiliary feedwater pumps and battery or diesel generator failure. In the third sequence $T_1M\bar{P}QLD_1$, which contributes about three quarters of the three sequence combined core melt probability, it is found that most of the contribution comes from the same electrical power failures that appear in the sequence $T_1M\bar{P}QL$. Therefore, electrical support system failures are controlling the failures of auxiliary feedwater and high pressure injection. Similar situations existed in the cut sets for the other sequences involved in the feed and bleed situation when common mode failures are essentially limited to electrical systems.

This final analysis, however, explicitly accounts for potential common mode failures of pumps and valves in the various systems. In this instance it is instructive to consider the sequence $T_2M\bar{P}\bar{Q}L$, which contributes about 13% of the overall probability of a core melt when feed and bleed is not available but secondary blowdown is available. If feed and bleed is viable, sequence $T_2M\bar{P}\bar{Q}L$ is replaced by two sequences, $T_2M\bar{P}\bar{Q}L\bar{E}H_1$ and $T_2M\bar{P}\bar{Q}L\bar{E}$, which have a combined contribution of about 1.3% of the probability of core melt. It should be noted that when feed and bleed is available, a large percentage of the probability of core melt is shifted to sequences which are unaffected by a feed and bleed capability. (Without feed and bleed, 88%, or 4.26E-5 out of 4.83E-5 is contributed by sequences which, when feed and bleed is available, are replaced by sequences that only contribute about 61%, 8.73E-6 out of 1.44E-5. See Table 10.1.)

In the sequence $T_2M\bar{P}\bar{Q}L$ the common mode failures related to auxiliary feedwater pumps contribute approximately 91% of the probability of core melt (5.82E-6 out of 6.38E-6) attributable to the sequence. Another 2.4% is related to common mode failure of the motor operated valves, while 1.2% is related to battery common mode failure and 0.9% is related to common mode failure of the AFW actuation system (see Table 10.2). As noted earlier, this is a significant shift from the original analysis which did not include common mode.

In sequences $T_2M\bar{P}\bar{Q}L\bar{E}H_1$ and $T_2M\bar{P}\bar{Q}LE$ (which replace $T_2M\bar{P}\bar{Q}L$ when F&B is available) the common mode failures of auxiliary feedwater again play a dominant role. However, because feed and bleed is available, the failure of auxiliary feedwater is always coupled with at least one other event. Sequence $T_2M\bar{P}\bar{Q}L\bar{E}H_1$, which involves failure of high pressure recirculation, is a negligible contributor and will not be analyzed in detail. However, in sequence $T_2M\bar{P}\bar{Q}LE$ the AFW common mode failure appears in 10 of the 15 cut sets. These cut sets contribute about 60% of the core melt probability related to this sequence (see Table 10.3). The battery common mode cut set essentially contributes the bulk of the remaining 40% of the total sequence core melt probability. Again we see a change in the nature of the failure mechanisms from those in the analysis without mechanical common mode failures

Table 10.1 Contribution by Probability of Core Melt of Accident Sequences Affected by Feed and Bleed*

Anal. No.	Sequence	Internal Event Core Melt Prob. w/o F&B	Contribution to Total Internal Event Core Melt Prob.(%)	Internal Event Core Melt Prob. w/F&B	Contribution to Total Internal Event Core Melt Prob.(%) of Replaced Sequences
72	S_2ML	4.0E-6	8.28	2.6E-8	.18
73	$T_1M\bar{P}QL$	5.2E-7	1.08	3.7E-7	2.57
74	$T_2M\bar{P}QL$	6.6E-7	1.37	2.8E-8	.19
75	$T_3M\bar{P}QL$	4.6E-7	.95	-	-
76	T_3QL	3.3E-7	.68	4.6E-9	.03
77	$T_4M\bar{P}QL$	5.4E-9	.01	-	-
78	$T_5M\bar{P}QL$	1.3E-9	.00	-	-
79	$T_1M\bar{P}\bar{Q}L$	1.5E-5	31.06	7.9E-6	54.86
80	$T_2M\bar{P}\bar{Q}L$	6.4E-6	13.25	1.9E-7	1.32
81	$T_3M\bar{P}\bar{Q}L$	1.5E-5	31.06	-	-
82	$T_4M\bar{P}\bar{Q}L$	1.7E-7	.35	1.7E-7	1.18
83	$T_5M\bar{P}\bar{Q}L$	4.1E-8	.08	4.1E-8	.28
		Σ 4.26E-5	88.17	Σ 8.73E-6	60.61
Total Core Melt (internal event w/o F&B)**				4.83E-5	Δ = 3.39E-5
Total Core Melt (internal event w/F&B)				1.44E-5	

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* See Tables B.16 through B.19 for details

**Credit for Secondary Blowdown is given here

Table 10.2 Selected Cut Sets for Sequence $T_2M\bar{P}\bar{Q}L$ (Sequence Applies When Feed and Bleed Not Available)

Core Melt Probability	Multiplier		Cut Sets
5.82E-6	$T_2M\bar{P}\bar{Q}$	*	AFW-PUMP-CM
1.55E-7	$T_2M\bar{P}\bar{Q}$	*	AFW-3VALVE-CM * AFW1C-PTS-LF
7.71E-8	$T_2M\bar{P}\bar{Q}$	*	AFW1C-PTS-LF * AFW1A-PMS-LF * AFW1B-PMS-LF
7.68E-8	$T_2M\bar{P}\bar{Q}$	*	LOSS-OSP * BATTERY-CM
5.82E-8	$T_2M\bar{P}\bar{Q}$	*	AFASLOG-UTM-CM
1.97E-7	$T_2M\bar{P}\bar{Q}$	*	71 Additional Cut Sets Involving Only AFWS and Valves

Σ 6.38E-6

AFW-PUMP-CM Contribution 5.82E-6/6.38E-6 = .91

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0164

Table 10.3 Cut Sets for Sequence $T_2M\bar{P}\bar{Q}$ LE (Sequence Applies When Feed and Bleed Available)

Core Melt Probability	Multiplier		Cut Sets
7.68E-8	$T_2M\bar{\bar{P}}\bar{Q}$	*	LOSS-OSP * BATTERY-CM
4.66E-8	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFW-PUMP-CM * BLKVLV-1403-LF
4.66E-8	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFW-PUMP-CM * BLKVLV-1405-LF
1.75E-8	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFW-PUMP-CM * OPF-MACT-FNB
1.24E-9	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFW-3VALVE-CM * AFW1C-PTS-LF * BLKVLV-1403-LF
1.24E-9	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFW-3VALVE-CM * AFW1C-PTS-LF * BLKVLV-1405-LF
4.66E-10	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFASLOG-UTM-CM * BLKVLV-1403-LF
4.66E-10	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFASLOG-UTM-CM * BLKVLV-1405-LF
1.75E-10	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFASLOG-UTM-CM * OPF-MACT-FNB
6.98E-11	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFAS-SENS-CM * BLKVLV-1403-LF
6.98E-11	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFAS-SENS-CM * BLKVLV-1405-LF
5.82E-11	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFASACT-A-LF * AFASACT-B-LF * BLKVLV-1403-LF
5.82E-11	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFASACT-A-LF * AFASACT-B-LF * BLKVLV-1405-LF
3.10E-11	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFW1C-PTS-LF * AFW9104-XOC-UTM * BLKVLV-1403-LF
3.10E-11	$T_2M\bar{\bar{P}}\bar{Q}$	*	AFW1C-PTS-LF * AFW9104-XOC-UTM * BLKVLV-1405-LF

Σ 1.91E-7

0165

where failures in emergency electric power systems played a dominant role. These two sequences contribute only about 1.32% of the total internal event core melt probability when feed and bleed is available or 1.9E-7 out of 1.44E-5.

Under these circumstances it is clear that any modifications which are proposed to improve auxiliary feedwater reliability (such as an independent diesel driven pump or an independent add-on train of auxiliary feedwater) should improve the situation whether or not feed and bleed is available. If feed and bleed is not viable (or not pursued due to other considerations as discussed below) then modifications which reduce or negate the impact of auxiliary feedwater system failures become even more significant.

An additional issue related to feed and bleed has been examined. As described elsewhere, the base case analysis for St. Lucie assumes feed and bleed procedures are available and, therefore, it is a viable mode of operation for purposes of this analysis. It was assumed originally that the PORV block valves would be closed to prevent PORV leakage. The influence of this assumption was examined by conducting an analysis with the block valves open. For the twenty-four F&B sequences affected by the block valves, the contribution to core melt probability is 1.13E-4 with valves closed and 8.09E-5 with valves open. This comparison is for sequences to which recovery has not been applied. Under these conditions the probability of core melt attributable to these sequences declines by 3.2E-5 or about 28% with the valves open (see Table B.21). Table B.22 presents these sequences after recovery has been applied. The results are as follows:

Total p(cm) for these sequences w/rec & F&B, Blk Val closed	8.30E-6
Total p(cm) for these sequences w/rec & F&B, Blk Val open	7.24E-6

or about a 13% reduction. Therefore, given the overall uncertainty in the analysis, the operating mode for the block valves has a small effect upon the probability of core melt.

In this investigation the addition of a dedicated feed and bleed system was not pursued for several reasons. First, if a dedicated system were added which was not completely independent (i.e., connections to RCS in all new piping with no reliance on existing valves, etc.), then it would be subject to failures such as operator errors and common mode failures of support systems. If a dedicated system without a separate dedicated power supply system were included, then it could still be subject to the failures in the emergency power system. This was illustrated in the results of the System 80 PORV Study.[17] In that instance there was no benefit for response to LOSP

0166

transients. There was a benefit for SGTR and SBLOCA events because these events were dominated by local HPI failures which are countered by the added redundancy of a dedicated F&B system. Even so, the overall reduction in frequency of core melt in tha study was only about a factor of 2.5. At St. Lucie the small break accident sequences to which F&B applies involve failures after a transient initiating event and so one would expect only limited, if any, improvement even with a dedicated system. (NOTE: In this TAP A-45 study tube rupture in a single steam generator is considered isolatable and therefore a transient initiating event. Simultaneous tube ruptures in both steam generators were not considered.) Second, the benefits of a completely independent feed and bleed system were not evaluated in the System 80 PORV Study.[17] However, the costs associated with such a dedicated, independent system are substantial based upon what was done in the earlier study. A rough extrapolation of the available information suggests that an additional pump and associated hardware would cost in excess of $6 million if the equipment were located in an existing structure. It was also estimated that a dedicated emergency electrical power system could run $11 million. (This is consistent with current TAP A-45 estimates of costs associated with adding an emergency diesel generator at another site.) There would be substantial added construction costs if the dedicated pump were housed in a new separate structure. All of this would make the cost of a fully independent feed and bleed system 3 to 4 times that of other viable alternatives and a significant fraction of those for the add-on auxiliary feedwater/makeup system which removes decay heat without contamination of containment. Third, the ADHR study results[11] indicate only about a factor of 2.5 improvement in core melt probability with an add-on HPI system that has a dedicated power supply. The add-on HPI was not as effective as an add-on AFW in reducing core melt probability at comparable cost. The add-on AFW system gave about a factor of 20 reduction in core melt probability. Given these earlier results and some of the timing and thermal hydraulic considerations outlined below, a dedicated F&B system was not examined for St. Lucie. However, while this report was in progress, two concepts for a feed and bleed system were proposed by G. A. Reed of the Advisory Committee on Reactor Safeguards.[19] These concepts will be examined in the overall context of TAP A-45 to evaluate their feasibility, costs and effect upon core melt probability and risk. Initial estimates are that p(cm) might be reduced by factors of 10 to 20, depending upon the configuration adopted, but impact estimates are not yet available.

10.1.2 Impacts Associated with Feed and Bleed

In a situation such as that which exists at St. Lucie where feed and bleed can be used without installing new equipment, there are no direct capital costs. However, there are other impacts or implications of feed and bleed which need to be considered.

Thermal-hydraulic Phenomenology - Because of the broad interest in feed and bleed as a last resort decay heat removal method, questions have been raised regarding its actual viability in terms of the thermal-hydraulic phenomenology and potential environmental effects on equipment. Los Alamos National Laboratory (LANL) has examined the phenomenological questions very extensively under NRC sponsorship. The LANL results have been reported in a series of draft reports (References 13 through 15). The objectives of the LANL work were to:

1) Evaluate success or failure of feed and bleed in specific reactors and for specific accident sequences.

2) Identify plant specific and generic insights.

3) Use insights to predict feasibility of feed and bleed for PWRs for which detailed analyses were not performed.

In conducting their investigations LANL used audited models of three specific plants. In this instance audited implied that all data is traceable to its source and that the input values are generally accepted as being realistic.

Specific response calculations were made by LANL for Calvert Cliffs 1, Zion 1, Oconee 1 and H. B. Robinson 2 using the TRAC-PF1 thermal-hydraulics code. It is also important to bear in mind the underlying assumptions and success criteria used by LANL. In References 13 and 14 LANL assumed that the plant equipment needed for F&B was available and fully operable throughout the transient, that is, equipment environmental qualification issues were not considered; nominal performance was used. In Reference 13 LANL further assumed that individual items of equipment were not functional, in one instance only 1 of 2 PORVs was used, in another only 1 of 2 HPI pumps. In those instances where the plant was being taken from reactor trip to hot standby, success was defined as attainment of stable RCS pressure above LPI and accumulator actuation levels, vessel mass inventory not decreasing and cladding temperature below or near saturation. When going from reactor trip to hot shutdown, success was defined as controlled RCS cooling and depressurization to LPI or RHR cut-in conditions.

A wide range of transient conditions was investigated by LANL including loss of feedwater, loss of offsite power, main steam line breaks, main feed line breaks, and steam generator tube rupture. These are described in detail in the referenced draft reports.

LANL concluded from this work that for the plants studied in detail, under the assumptions stated, feed and bleed would be successful if initiated no later than the time of steam generator dryout although the operating boundaries shift when less than nominal equipment is used. The actual time under

0168

nominal conditions varies from about 8 minutes for Oconee 1 (low inventory, once through B&W steam generator) to about 40 minutes (large U-tube Westinghouse steam generators). Because St. Lucie is a Combustion-Engineering 2-loop unit very similar to Calvert Cliffs (which was the unit modelled by LANL) it is appropriate to consider the LANL results for Calvert Cliffs in somewhat more detail. In this regard the following has been adapted from Reference 32. The feed and bleed transients investigated for Calvert Cliffs are uniquely distinguished by the PORV and safety injection delivery capacity. The PORV capacity is intermediate between that for Zion (larger) and Oconee (smaller). For LOSP and LOFW transients, primary pressure continued to increase for a short period of time after the PORVs and SRVs opened. Above the HPI shutoff head (1275 psia) only charging flow from constant displacement pumps is available.

The following conclusions about feed and bleed were reached:

1. To bound all transients, safety injection must be initiated by steam generator secondary side dryout.

2. If safety injection is not manually initiated, auto-initiation at containment overpressure may be effective in preventing gross core damage, but core void will occur.

3. If nominal equipment is not available, feed and bleed is unsuccessful if initiated at steam generator dryout.

4. If feed and bleed is initiated prior to steam generator dryout, hot shutdown conditions can be achieved.

5. If feed and bleed is initiated after steam generator dryout, the PORV should not be locked open until after a subcooling margin has been established with injection.

LANL did not have an opportunity in the course of this study to analyze St. Lucie. However, they have made an extrapolation of their results to such plants. Therefore, 1) on comparing the key parameters for Calvert Cliffs and St. Lucie in Table 10.4, and 2) considering that feed and bleed procedures have been adopted, it is reasonable to conclude that feed and bleed can be successfully conducted from the phenomenological point of view.

Table 10.4 Selected Combustion-Engineering Plant Parameter

Parameters	Calvert Cliffs	St. Lucie
Power (MW_t)	2,570	2,560
SG Inventory (kg)	124,700	~ 60,781
PORV Capacity		
(#/hr/MW_t)	56.7	57
(#/hr)	145,700	145,900
HPI Capacity		
gpm @ 1000 psi	410	435
gpm @ 1600 psi	0	0
Positive Displacement Pump (gpm)	132	132
SG Dryout Time (min)	16	16

In summary, these analyses indicate that feed and bleed can be successful in Combustion Engineering plants provided the process is initiated early enough. The initiation time can be as little as 10-15 minutes into an event depending upon when reactor scram occurs. It would appear that this puts a considerable pressure upon the operators. A decision to open the primary must be made before there is really any time to establish reasons for other equipment not functioning. It is noted that one of the concepts proposed by Reed[19] alleviates this concern by blowing down to, and recirculating from, a closed system.

10.1.3 Operational Issues

Another aspect of the application of feed and bleed must also be considered. Some owners group guidelines direct the operator to attempt to restore feedwater to the steam generators until primary system pressurization and coolant heatup begins. Once the primary system pressure and temperature begin to increase following loss of the secondary heat sink, the operator is directed to quickly establish once through cooling using a feed and bleed procedure. However, there has always been some question about the willingness of operators to open the primary system at early times given the history of education which says maintain primary system integrity. This concern seems to be borne out based upon recent reports on the June 9, 1985, incident at Davis-Besse, in which operators may have delayed initiating feed and bleed.[34]

10.1.4 Environmental Qualification Concerns

In addition to the questions related to the thermal-hydraulic phenomenology, there are concerns about equipment qualification, that is, will the necessary equipment function under the environments that may be created.

Using the LANL calculations to define the mass and energy released into containment, pressure/temperature profiles were

0170

developed by the NRC staff as a function of containment system performance. A large dry containment was used for the calculations. The general findings of that study as quoted in Reference 21 are:

"Following a feed and bleed transient if all the active containment heat removal systems fail, the containment pressure and temperature could exceed the design values, but would level off before the containment pressure reaches the ultimate containment strength of more than two times of design pressure. The containment pressure and temperature conditions following a feed and bleed transient are lower at the peak and at some later time (10^4 sec) than those following a LOCA. The conditions from feed and bleed could be slightly higher than those from LOCA during a period of several thousand seconds and at very later time (10^4 sec) when the elevated pressure and temperature are not significantly higher than those during normal operating conditions. The implications of this period to equipment qualification, during which the containment pressure and temperature conditions of feed and bleed are slightly higher than those from LOCA, is plant dependent and instrument specific."

At this point in time the qualification profiles for specific pieces of equipment have not been examined, and there are some unanswered questions related to the differing pressure-time and temperature-time profiles. Those of the LOCA environments peak rapidly and then drop off,[22] while the F&B environments, although lower in peak amplitude, exist for longer periods of time. The effect upon equipment performance has not been evaluated.

10.1.5 Summary

The preceeding analysis clearly shows that a feed and bleed capability can have a significant effect upon the probability of core melt. Also, the thermal-hydraulic studies certainly suggest that feed and bleed is possible from a phenomenological point of view. However, these apparent benefits must be viewed in the light of: a) the rather stringent time constraints revealed, as decisions to feed and bleed must be made early in the accident progression for it to be successful; and b) the apparent operator reluctance to "open" the primary system. These latter concerns may be alleviated by a closed system such as has been proposed by Reed.

10.2 Secondary Side Blowdown

In this St. Lucie analysis, the term secondary blowdown, is used to describe the use of the atmospheric dump valves to rapidly depressurize the secondary side of the steam generators. This action is accompanied by primary side depressurization due to the increased primary to secondary heat transfer and

0171

subsequent cooling of the primary. The goal is to reach low pressure injection set points or residual heat removal cut-in conditions so that the system can be put on closed cycle cooling.

There has been some discussion related to the use of pressurizer sprays in conjunction with the rapid secondary depressurization to depressurize the primary after a loss of offsite power transient. In addition to an examination of the available literature on secondary side blowdown, Los Alamos National Laboratory (LANL) was asked to make some specific calculations. The LANL results are discussed here along with some insights from the probabilistic assessment.

10.2.1 Value of Secondary Side Blowdown

In the internal event analysis of the St. Lucie Nuclear Plant, the event trees were first constructed to include the use of secondary side blowdown (see Section 2 and Appendix B) and then rerun without this capability. The success criteria included the use of both the atmospheric dump valves and the pressurizer sprays in this process. As shown in Appendix B, the use of secondary side blowdown influences 21 potential accident sequences. If a secondary side blowdown capability did not exist, then these 21 sequences reduce to 7 sequences. The specifics of the sequences replaced are discussed in Appendix B, but some comments are included here. The combined core melt probabilities for the affected sequences, prior to applying recovery actions to these sequences are:

P(cm) (affected sequences without blowdown)	4.34E-5
P(cm) (affected sequences with blowdown)	2.82E-5

After recovery is applied to these sequences, the values are

p(cm) (affected sequences without blowdown)	6.26E-6
p(cm) (affected sequences with blowdown)	3.48E-6

On this basis the secondary side blowdown capability reduces the estimated core melt probability at St. Lucie by 2.8E-6. This is a much smaller effect than that attributed to the feed and bleed capability (see Section 10.1.1 above). However, if secondary side blowdown and pressurizer sprays are not available the probability of core melt p(cm) (internal events only, but with feed and bleed), would be about 1.71E-5. If neither feed and bleed or secondary depressurization is available the p(cm) rises to about 5.4E-5.

In the case of feed and bleed the overall gain (3.4E-5) was significant because the required systems with or without feed and bleed had quite different vulnerabilities. The improvement with secondary side blowdown is less dramatic (~ 3E-6) because differences in the required support systems with and

0172

without blowdown are not as pronounced. This can be illustrated by considering the SBLOCA sequence $S_2M\bar{L}D_1$ which contributes about 54% of the core melt probability for the affected sequences without blowdown cited above. If secondary side blowdown is feasible, there are three possible sequences; $S_2M\bar{L}D_1H_2'$, $S_2M\bar{L}D_{12}$ and $S_2M\bar{L}XD_1$. In all of these sequences the initiation of the SBLOCA (S_2) is accompanied by loss of main feedwater (M) but auxiliary feedwater is available ($\bar{L}$). In the first instance the probability of core melt is influenced by the failure probability of high pressure injection (D_1). In the latter three sequences the combined probability of failure of: high pressure injection (D_1) and low pressure recirculation (H_2'); high pressure and low pressure injection (D_{12}); and secondary side blowdown (X) and high pressure injection (D_1), respectively, are the issues of concern. Sequences $S_2M\bar{L}D_1H_2'$ and $S_2M\bar{L}XD_1$ are negligible constributors while $S_2M\bar{L}D_{12}$ contributes 100% of the combined probability. Therefore, the principal comparison is between sequences $S_2M\bar{L}D_{12}$ and $S_2M\bar{L}D_1$. In sequence $S_2M\bar{L}D_{12}$, given the initial conditions $S_2M\bar{L}$, two injection systems must fail to cause core melt, while in sequence $S_2M\bar{L}D_1$ failure of only the high pressure system is enough to cause core melt. These two sequences have virtually the same cut sets since the failures are dominated by failure of the safety injection signal actuation or logic, service water and component cooling water valve failures, or common mode pump failures of the support systems. Therefore, the secondary blowdown capability does not improve the core melt probability since both the high and low pressure injection support systems have failed. Therefore, having a secondary blowdown capability does improve the core melt probability by about 5.2% whereas the feed and bleed improvement was about 63% if no credit if given for secondary blowdown, and 66% if secondary blowdown credit is given.

10.2.2 Systems Required for Successful Depressurization

Several years ago EG&G (INEL) investigated small break accident mitigation at Zion 1, assuming coincident failure of both charging and high pressure injection.[23] The conclusions of that study were:

"The largest break diameter which will prevent a depressurization to the accumulator injection pressure is about 6 cm (2.4 in).

"The small break sequence with coincident failure of charging and HPI rapidly leads to a core uncovering but sequence timing is directly affected by break diameter.

"The opening of a single atmospheric dump valve is not effective in preventing core uncovering.

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"The opening of all four ADVs at 10 minutes has been shown to be an effective means of preventing core uncovering following 2.54 cm (1 in) and 5.08 cm (2 in) diameter breaks."

In these INEL investigations for Zion 1 no pressurizer sprays were used. The atmospheric dump valves had a flow area of 0.01015 m^2 (0.1092 ft^2) each, sufficient to pass 112.1 kg/S (247 #/s) of saturated steam at an upstream pressure of 7.72 MPa (1120 psia).

Similar calculations were performed by Combustion-Engineering (CE) to address the question of how CE System 80 plants handle small break LOCAs without PORVs.[24] The calculations were performed for a 3410 MWt class plant. Aggressive secondary side cool down at 100°F/hr was initiated 15 minutes into the event. Results indicated that the Safety Injection Tanks (SIT) begin to inject at about 3500 seconds. SIT injection continues until the RCS pressure reaches 100 psi. The low pressure injection pumps (LPI) will begin to inject at 200 psi. Based upon inspection of the graphical results in Reference 36, LPI operation should begin between 6000-7000 seconds. The ADVs used had a capacity of 195.2 #/s at 900 psia with an effective flow area per valve of 0.108 ft_2. There was one ADV per steam generator.

J. D. Harris of the Nuclear Operations Analysis Center at the Oak Ridge National Laboratory examined atmospheric steam dump systems for a number of PWRs.[25] The specific initiating events of concern are not spelled out in the Harris study, but the text implies that LOCAs and transients are considered. Harris does specifically state that he is concerned about heat removal after Main Steam Isolation Valve closure after some Design Basis Event. Based upon his relatively simple analysis of 13 PWRs, Harris concluded that: 1) three plants (Indian Point 2 and San Onofre 2 & 3) could achieve RHR entry using available auxiliary feedwater; 2) four plants (Fort Calhoun, Calvert Cliffs 1 & 2, and Davis Besse 1) could not and; 3) six (Palisades, Salem 1 & 2, Crystal River 3, Beaver Valley 1, Millstone 2, and McGuire 1) could not if there were steam dump valve failures. Thus, the Harris approach agrees with the CE analysis that the 3410 MWt class can depressurize. Although the previous estimates by both INEL and CE suggest that existing ADVs are adequate to reach RHR conditions, the questions raised by Harris prompted further examination.

Los Alamos National Laboratory (LANL) was asked to examine the question using the TRAC/PF1 thermal-hydraulics code. The Calvert Cliffs plant model was used for this examination for several reasons. One, Calvert Cliffs was one of the plants Harris indicated could not reach RHR cut in. Two, this plant model has been used in several prior studies. Three, Calvert Cliffs was the reference plant for the cold shutdown investigations in TAP A-45.[26] LANL examined the use of ADVs

0174

in response to a small break LOCA with failure of high pressure injection[27] and to a loss of offsite power transient.[28] The major conclusions from the LANL work are as follows. From Reference 27 for the small break LOCAs:

"The primary system will cool and depressurize to the low-pressure injection system operating pressure and design temperature limits using only water supplies from the safety-grade condensate water storage tank.

"The procedure is successful even with additional equipment failures such as the failure of one atmospheric dump valve to open or the safety injection tanks to discharge. Such failures lengthen the time to start the low-pressure injection system.

"Because the depressurization rate is enhanced by the small break, the cooldown rate defines the time to low-pressure injection system operation.

"Doubling the auxiliary feedwater flow increased the cooldown rate but the increase was small to moderate.

"No calculation was extended to a low-pressure injection system operating temperature of 422 K (300°F). The cooldown from the design temperature limit to the operating temperature was very slow. Because the secondary-side pressure had reached a quasi-steady state it seems that further cooldown would track the decrease in decay power."

The LANL results for SBLOCA at Calvert Cliffs are consistent with the INEL Zion 1 study.[23] The depressurization of the primary is "assisted" by flow out of the break and there is sufficient inventory available to adequately cool down. It would appear that a similar situation should exist at St. Lucie, but detailed calculations have not been performed.

Similarly, from Reference 28 for the LOSP transient:

"Cooldown and depressurization to RHR entry conditions does not occur for the case where operator uses only existing plant ADVs and AFW.

"If the operator uses APS in addition to existing ADVs, depressurization is improved, but RHR entry is still unattainable, because of secondary side cooldown and depressurization limitations.

"Doubling the ADV capacity still may not provide sufficient flow capability to bring secondary pressure and temperature down low enough to allow primary cooldown to RHR entry conditions.

"By increasing the ADV capacity to 400% [of nominal] the plant can reach RHR entry conditions within 3-1/2 hours from the initiating event. A 500% flow reduces this entry time to about 3 hours."

With regard to the LOSP transient, after it was ascertained by LANL that the APS was required for depressurization using only one ADV per steam generator, the APS was used in all subsequent calculations. Therefore, the ability to depressurize Calvert Cliffs with larger relief capacity but without APS was not addressed. However, to put some perspective on this question, it is noted that the St. Lucie LOSP sequences that can be affected by the secondary blowdown capability contribute only 6% of the core melt probability discussed in Section 10.2.1 above. This suggests that the response to the LOSP transient is not a major issue. It is our understanding from discussions with the St. Lucie staff that they could reach cold-shutdown with existing CST inventory given a LOSP.

10.2.3 Summary

In summary, it appears reasonable to state that: 1) the secondary side blowdown capability only has a demonstrable effect upon the probability of core melt; 2) given SBLOCA events with failure of HPI, aggressive cooldown using the ADVs will reduce the primary pressure to RHR entry conditions; and 3) for a LOSP transient, RHR entry conditions may be reached using a combination of APS and ADVs, but the situation is very plant specific. Successful cooldown to RHR entry conditions using only ADVs after a LOSP transient has not been fully explored, but in the case of St. Lucie this sequence is a minor contributor to the core melt probability.

11.0 SUMMARY, CONCLUSIONS, AND OBSERVATIONS

This report documents the investigation of the adequacy of decay heat removal (DHR) at the St. Lucie Nuclear Plant. This particular plant was selected as an example plant for several reasons. One, the results of an initial qualitative screening process suggested that there were some potential DHR vulnerabilities that deserved further study. Two, it is "typical" of the layout and construction practices of a number of plants of similar vintage. Three, the utility management, Florida Power and Light, expressed a willingness to participate in this aspect of the USI A-45 investigations. The following sections highlight some of the findings and insights including non-quantifiables, discuss aspects of DHR issues not addressed specifically, and finally make some generic observations.

11.1 The DHR Assessments

This program was not designed to conduct full-scope probabilistic risk assessments (PRA) of the plants being studied. However, it was established very early in the program that limited-scope assessments would be necessary to adequately address the issues. Because TAP A-45 is considering the full range of potential failure mechanisms, the study includes a PRA based examination of internal random and systematic faults and a simplified PRA based examination of special emergencies. This study uses the term "special emergencies" to describe those events such as fires, floods, winds, etc., which are often labeled external events elsewhere. These latter investigations are characterized as "simplified" because many of the analysis techniques are still under development and therefore, in some instances, have substantial subjective engineering judgment included.

11.1.1 The Internal Analysis

The details of the internal analysis are presented in Section 2 and Appendices A and B. From this analysis it was established that the random internal failures (including recovery actions) had an associated core melt probability of 1.4E-5. There were 10 internal vulnerabilities identified, the effects of which accounted for 56% (8.1E-6) of the core melt probability. No equipment or system modifications were proposed to correct or reduce any of these vulnerabilities.

11.1.2 The Special Emergency Analysis

The details of the special emergency analyses are presented in Section 3 and Appendices C through I. The special emergencies included in this study are: seismic events, fire, internal and external floods, extreme winds, lightning and internal sabotage.

The seismic contribution to core melt was estimated to be 1.33E-5, 95% (1.26E-5) of which was contributed primarily by

.177

three specific vulnerabilities. The fire contribution to the core melt probability was 4.38E-5 attributable to one vulnerability. Internal flooding and extreme wind were insignificant contributors, while external flooding had a probability of 3.2E-6. Lightning contributed 2.0E-7 while the sabotage analysis was not quantified, but the add-on decay heat removal system would be helpful in countering potential sabotage.

11.1.3 Assumptions and Bases for the Analyses

It seemed appropriate to assemble in one place for reference the assumptions or other special factors upon which this case study is based. This is done here under six headings: internal analysis, external analysis, dedicated add-on system, impact analysis, consequence analysis, and value/impact analysis. While these lists may not be complete, they should describe most of the important bases for the analyses that may affect the results.

11.1.3.1 Internal Analysis Assumptions/Conditions

Generic

1. System configurations and equipment locations were derived from the FSAR, drawings supplied by the utility, plant visits, and communications with plant personnel.

2. Operational procedures in general were not analyzed in detail and we relied upon statements from plant personnel regarding procedures.

3. The scope of initiating events was limited to small LOCA and transients including loss of an AC bus or a DC bus but excluding the loss of service water initiator. ATWS was not considered.

4. System success criteria used are generally conservative.

5. Containment failure and sump flashing do not result in ECCS failure due to pump cavitation.

6. Auxiliary Feedwater System (AFWS) success is sufficient for core cooling.

7. Feed and Bleed is a viable mode of operation and no containment overpressure protection is required in that mode.

8. Even if the pressurizer safety relief valves are not required to provide RCS overpressure protection, these valves have a 7% chance of opening anyway.

9. The frequency of loss of offsite power quoted in NUREG/CR-2728, i.e., 8.6E-2 per reactor year, was used

rather than other suggested generic or plant specific numbers. This provides a consistency between analyses.

10. Various component failure probabilities were derived in these analyses and may differ from other data bases.

11. Circuit breaker failures were grouped with the associated buses for simplicity.

12. Common mode pump and valve failure probabilities were derived based upon ASEP information.

13. Control circuit failures were combined with the local fault probabilities for pumps and motor operated valves.

14. Recovery of the PCS was only permitted for selected accident sequences depending on the availability of power and whether the initial loss occurred as part of the initiating event or subsequent to it.

15. Only one recovery action was allowed in addition to recovery of LOSP or PCS, except when the allowable recovery time is greater than 2 hours, 2 recovery actions were permitted.

16. Cut sets with two UTM events from the same system were considered invalid.

17. Hardware failures of pumps and valves were not considered recoverable except for that part of the failure attributed to control circuit failures for valves which did have a recovery applied.

18. Normally-closed motor-operated valves in pump suction lines were considered recoverable since it was assumed operators would shut down the corresponding pumps before cavitation occurs based on discharge pressure and flow indication, thus making the recovery meaningful.

19. Actuation system failures are assumed to be remotely recoverable by manual initiation of the particular system.

20. Successful safety injection extends any accident sequence time to core melt by approximately 3 hours.

21. Recovery was only applied to the top cut sets of the top accident sequences--approximately 95%

22. The Power Operated Relief Valve (PORV) was assumed to be blocked.

23. Reactor coolant pump cooling was not modeled and, therefore, it is assumed that the seals do not require cooling.

0179

Specific

1. Any valves and pumps that were actuated for injection will still need actuation and power for the recirculation phase in case the injection was not required or pumps were turned off prior to the recirculation phase.

2. In order to avoid a fault tree loop where the diesel generator depends on DC power which, in turn, depends on the diesel generator, it is assumed that only component faults of the applicable DC power train result in failure of the corresponding diesel generator.

3. Simultaneous shutdown of both units is not considered relative to systems shared between units. Thus all swing safeguard equipment can be used by the unit being shut down.

4. Reactor coolant pump seal failure is not considered since this is an issue which, if determined to be important, will be solved by another program. Thus neither CVCS or CCWS cooling of seals is modeled.

5. The CVCS is not needed for makeup while the pressure is being reduced for the safety injections pumps to be effective since the AFWS is used to speed up this process. The PORVs would not normally be used to get the pressure down and the pressurizer sprays are not required although they could be used.

6. The PORVs will not be used to increase the depressurization rate for secondary blowdown due to the concern that the core could be uncovered before the LPIS can inject water into the primary system.

7. The CVCS would be needed on transients eventually for makeup and for the reactor coolant pump seals but only for cold shutdown which is not covered by this analysis.

8. In the event L defined as the AFWS and secondary steam relief, the secondary steam relief is not fault tree modeled since there is adequate redundancy, i.e., ADVs, or safety relief valves, or the steam dump to the condenser given condenser vacuum is available except in the case of a double steam generator tube rupture which is not being addressed.

9. Containment failure does not imply core melt due to containment sump flashing as has been assumed in many past PRAs. This permits the containment systems to be shown in a separate event tree.

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10. The AFWS is sufficient to remove all decay heat without the RHR heat exchangers during the shutdown heat removal phase, i.e., to hot shutdown.

11. The emergency batteries have an 8 hr rating without a battery charger.

12. CCW system has a third pump that can be manually aligned as a recovery action.

13. ICW system also has a third pump, not modeled, but available as a recovery action.

14. Manual recovery of the turbine driven AFW train is possible after a loss of DC power.

11.1.3.2 External Analysis

General

A number of external events (special emergencies) were excluded because they were beyond the scope and depth of our analysis to examine carefully or because of one of the following reasons.

a. The event is of equal or lesser damage potential than the event for which the plant has been designed.

b. The event has a significantly lower mean frequency of occurrence than other events with similar uncertainties.

c. The event cannot occur close enough to the plant to affect it.

d. The event is included in the definition of another event.

e. The event does not directly affect the decay heat removal system.

f. The event is slow in developing and there is sufficient time to go to cold shutdown.

This left seven events for further consideration for which the corresponding assumptions are given below.

Seismic

1. Seismic hazard curves were taken from other studies and not developed specifically for this program.

2. It was assumed that loss of DC fails the turbine driven emergency feedwater pump.

Fire

1. Only fires in a single room were considered.
2. Fire frequency and suppression reliability was based on available industry data.
3. Two transient combustible exposure fires were assumed to bound all transient and electrically initiated fires.
4. It is assumed that a fire occurs independently of any other initiating event and the operators will always scram the reactor once they know there is a fire.
5. Passive components will be assumed to be unaffected by a fire.
6. Offsite power is assumed to be available during the fire unless the fire is capable of causing a LOSP.
7. Fire growth, spread, and suppression were treated independently.
8. No core melt timing was considered in the analysis.
9. When exact locations of cables could not be accurately determined, their locations were assumed.
10. Credit was given in the calculations for radiant energy shields, conduit, solid bottom cable traps, and completely enclosed cable trays.

Internal Flood

1. Internal flood frequencies, component damage thresholds, and physical room considerations were based on previous industry wide data bases.
2. It is assumed that an internal flood occurs independently of any other initiating event and the operators will always scram the reactor once they know flooding is occurring.
3. Flood barriers always maintain their structural integrity.
4. Flooding via drains backing up into other areas was not considered.
5. Only single flood sources were analyzed.
6. Components fail from submergence in water to a critical level or from direct spray from the flood source. It is assumed that switchgear and buses will fail if they are

standing in six inches or more of water. An electric motor is assumed to fail when the water level reaches the bottom of the motor casing.

7. Offsite power is assumed to be available during internal flood when the flood is capable of causing an LOSP.

8. No core melt timing was considered in the analysis.

External Flood

1. It is assumed that loss of offsite power will occur during an external flood event.

2. Flooding due to rain or snow melt would be slow developing and thus not present any threat.

3. Equipment failure is assumed given submergence.

Extreme Wind

1. Equipment protected by concrete structures at least 12 inches thick are adequately protected against tornado missiles.

2. All openings were either protected by barrier walls, tornado missile barriers, or if less than 100 square feet were considered too small to be vulnerable due to a low mean frequency of impact.

3. Loss of offsite power is assumed in extreme winds that could damage the plant.

4. Recovery of the diesel generator is possible, given bent exhaust stacks, by cutting the stacks to allow air intake.

Lightning

1. Lightning protection equipment was not specifically evaluated as to proper installation or maintenance.

2. It is assumed offsite power fails with a likelihood of 0.5 when lightning strikes the plant.

3. A generic ground flash density for the southern U.S. was used in the analysis.

4. It was assumed from the literature review, site inspection, and discussions with plant personnel that the plant was designed according to NRC recommendations to be able to withstand a lightning strike of 200 kA.

0183

5. The St. Lucie site was estimated to be 1/4 square kilometer in area.

Sabotage

This is discussed in Appendix I under separate cover.

11.1.3.3 Dedicated Add-on Analysis

1. The system selected was based on the results of previous NRC studies, and the assumption that forced or natural circulation through the steam generators provides the most effective path for decay heat removal from the core.

2. Offsite power will be used if available but a completely independent power system is included.

3. All structures and connections are designed to seismic Category I specifications. All equipment is assumed to be designed to withstand all special emergencies.

4. Common mode between add-on trains was assumed negligible.

5. Pipe rupture in the add-on system was considered insignificant.

11.1.3.4 Impact analysis

1. All costs are in terms of 1985/1986 dollars.

2. Replacement power, if required, in order to implement an alternative is in 1985/1986 dollars.

3. In general, specific location for equipment and pipe runs were considered. In a few cases the space available may be uncertain.

4. Local labor rates were considered for all tasks.

11.1.3.5 Consequence Analysis

1. The mapping of accident sequences and containment system failures to release categories was accomplished using results of past studies adapted to TAP A-45 so no specific phenomenological computer runs were accomplished.

2. Containment failure mode probabilities were updated in light of recent studies to allow for a significant probability of containment success.

3. A range of offsite population doses were considered since the source term issue has not been settled.

4. No full scale uncertainty analysis was performed. Various issues are discussed in Section 8 considering the possible range of values for the results.

11.1.4 Base Case Results

When the results of the analyses are combined, the base case core melt probability is 7.44E-5 made up of the following contributors:

Internal	1.4E-5
Seismic	1.3E-5
Fire	4.4E-5
Internal Flood	ε
External Flood	3.2E-6
Extreme Wind	ε
Lightning	2.0E-7
	7.44E-5

11.1.5 The Alternatives and Revised Core Melt Probabilities

As noted above, specific modifications were recommended for the most significant vulnerabilities identified. These modifications were combined to define two possible alternatives which could be implemented at the plant (see Section 5.0). The impacts, costs, associated with implementing the alternatives were estimated using standard industry practices (see Section 6.0). The core melt probability was then reevaluated assuming the alternatives were in place (see Section 7.0) with the following results:

Alternative	Probability of Core Melt per Reactor Year	Change from Base Case
1	3.3E-5	4.1E-5
2	4.9E-6	7.0E-5

11.1.6 The Value-Impact Summary

The results of the core melt probability estimates were combined with the results of an analysis of containment systems performance and CRAC-2 calculations of offsite consequences to establish estimates of public risk (see Section 7.0). These estimates and the impact results were then combined in a value-impact analysis. Detailed results are tabulated and presented in Section 9.0. Table 11.1, which is the same as Table 9.6, summarizes the key results.

11.1.7 Discussion of Results

There are a number of conclusions or observations based upon Table 11.1 which should be highlighted.

0185

TABLE 11.1 ST. LUCIE - SUMMARY OF VALUE-IMPACT MEASURES (Central Value)

				V-I Analysis of Offsite Costs			V-I Analysis Based on Offsite and Onsite Costs		
Alternative No.	Change in Core Melt Probability	Offsite Averted Dose (p-rem)	Total Averted Dose Ratio	V-I Ratio	Net Benefit ($ $\times 10^{-6}$)	Dollars per p-rem	V-I Ratio	Net Benefit ($ $\times 10^{-6}$)	Dollars per p-rem
	Δp_m	V_2	ADR_n	VIR_o	NVB_o	DPR_o	VIR_n	NBV_n	DPR_n
1	4.14E-5	144	.66	.076	- 1.005	7556	- .279	.514	0
2	6.99E-5	228	.96	.002	-58.462	256987	.003	-55.903	190759

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a) There appear to be a few feasible modifications reflecting both internal and special emergency concerns which, if implemented, can significantly reduce the estimates of core melt probability, for example, those that are contained in alternative 1. This tends to confirm one of the original assumptions of the program, i.e., there are some DHR vulnerabilities in the existing plants and they can be countered.

b) The total averted population dose is on the order of hundreds of person-rems. However, the averted dose ratio indicates that the alternatives can be very effective in reducing population dose. For example, alternative 2 eliminates about 96% of the exposure predicted in the base case.

c) Considering only offsite costs, the value-impact ratio and net benefit measures both suggest that neither of the two alternatives are cost effective since they produce value-to-impact ratios less than 1 and negative net benefits. Also, the dollars per person-rem measures for these two alternatives are greater than the nominal $1000 per person-rem while the core melt probability is reduced by factors of 2 to 15.

d) If the value measures include the onsite costs that can be averted (personnel exposure, replacement power, cleanup, and loss of investment) then for alternative 1, all of the value measures would suggest that it is worth implementing. Since the net impact is a negative number, the value/impact ratio is also negative and is considered to be favorable.

e) In terms of the quantitative measures alternative 2, which includes the add-on independent train, does not fare well because it involves significant (tens of millions of dollars) capital expense. However, it does provide other non-quantifiable attributes (see Section 11.2) which must be considered.

11.2 The Add-on Decay Heat Removal System

11.2.1 Non-quantifiable Value Issues

As noted above, the add-on system capital expense puts the system at a disadvantage in a simple quantitative analysis. However, there are other attributes which must also be considered. The add-on system, because of its complete independence from other plant systems, will have a significant effect upon residual risk. It is unaffected by fires and/or floods which might occur in other plant buildings. Similar comments can be made about this system with respect to equipment qualification issues. The add-on is protected from steam line breaks and high to medium pressure line breaks which could occur in the plant. Furthermore, since it provides the independent backup to other systems, a variety of potential failures of normal safety systems are covered. Because it is housed in a separate structure and interfaces only with the primary system inside containment, the add-on provides additional protection against sabotage. Since all the equipment is in standby status and is a last resort backup to existing safety systems, added security (access control) is possible without a major impact upon routine day-to-day operations.

11.2.2 Comparison with European Practice

For simplicity in design and costing the details of the add-on system proposed here were extracted directly from prior work.[12] Therefore, it is a single train system. In the earlier work this approach was adopted based upon the philosophy that it is a "last resort" system, and therefore a number of redundant and diverse systems would have to be failed before it is used. In contrast, in a number of European installations of a similar nature redundant trains are used. However, their approach must be viewed within the general context of nuclear safety in Europe. In their "front line" safety systems they usually adopt the N+2 philosophy as compared to the N+1 approach in the United States. Therefore, they generally provide more redundancy. It should also be noted that in many instances their special emergency concerns are driven by the higher population densities near the sites, and the higher frequency of some events, e.g., aircraft crashes.

There is no doubt that increasing the redundancy would increase costs. Because time and resources originally led this program to the use of an existing design, cost data for a multiple train system is not available. These aspects will be considered in subsequent plant analyses for TAP A-45. Costs are also affected by the level of hardness selected for the structure housing the add-on system. The design here is for a Seismic 1 structure with limited access. Therefore, the add-on system will be available as long as the primary system it is designed to protect is available. This is generally consistent with the approach in Europe although there are instances there, notably in Switzerland, where the add-on is housed in a bunker with much greater hardness than the rest of the plant. Such bunkering may offer some advantage in coping with an external sabotage threat (which is not being considered in TAP A-45), but does not appear significantly more beneficial in coping with insiders than the system proposed here.

The design used here does not include provisions for putting the plant into the residual heat removal mode from the add-on system. Operators would have to take control of the usual plant systems either locally or from the control room. A number of the European plants do have provisions within the add-on system building for taking control of the normal plant systems and proceeding to cold shutdown. In several instances this was simply based upon the philosophy that if you are having problems, cold shutdown is the safest condition to be in.

Subsequent program investigations will examine the costs and design ramifications associated with adding such a capability.

11.3 Core Melts at Multi-Reactor Sites

11.3.1 Introduction

Several TAP A-45 plants, including St. Lucie, share their sites with a second reactor. These multi-reactor sites often share support systems such as electric power and auxiliary feedwater. We have found that many of the dominant cut sets computed for one unit will actually affect both units and in some cases simultaneous melts could occur. For example, cut sets involving an extended loss of offsite power could very well lead to core damage at both units. Additionally, modifications designed to benefit a single unit could have similar positive effects on the second unit.

The effects of accidents which could cause both units to melt have not been addressed explicitly in this study. In the baseline analyses, a single reactor has been analyzed and shared systems only modeled where necessary. Multi-reactor effects at St. Lucie have, however, been examined as a sensitivity issue. In our approach, estimates of the total probability of a single or dual unit core melt at a multi-reactor site, costs and effects of the proposed DHR modifications on both reactors and the consequences from accidents at a site have been used to estimate the value-impact of such site modifications.

From our observations, most two-unit melt scenarios occur after initiating events which more or less uniformly affect both units such as a total loss-of-offsite power and special emergencies such as earthquakes or floods. Therefore, we have assumed that all accidents initiated by losses-of-offsite power and special emergencies which affect the site are coupled so that a melt at one unit implies melt at the other. This is conservative since it is possible that offsite power might be lost at only one unit and not all site special emergencies would have the same effect on both units.

Therefore, the combined, two unit core melt probability at St. Lucie is estimated to equal the sum of the probabilities of the T1 accident sequences and site special emergencies calculated for one unit plus twice the probability of the other accident sequences (T2, T3, AC and DC bus initiators, and small LOCAs, fires, internal floods, and lightning) since these latter accidents are assumed to be independent at each unit unless an external event affects a common area (e.g., a shared cable spreading room).

The effects of the proposed DHR modifications on the total "site" accident probability are also estimated. Modifications

0189

which clearly affect only one unit and are equally applicable at the second unit are assumed to be added at both units. This doubles the cost of the modification and the resulting reduction in core melt frequency. For modifications which already clearly affect both units, no change in cost or amount of core melt reduction stemming from a modification is necessary. No change to the costs will be considered for modifications which may have some limited effect on the second unit. The amount of core melt probability reduction for these modifications will be examined and adjusted appropriately. Other associated costs of DHR modifications such as operations and maintenance costs, avertable power replacement costs, loss of investment costs, and site cleanup costs have been adjusted as deemed appropriate.

Given core melt at two reactors on a site, the radiological source term could double from what we have initially assumed. This will give rise to some increase in the offsite, 50 mile population dose. But, as discussed in Reference 35, the 50 mile population dose is relatively insensitive to the source term so these consequences will not double. Based upon an extrapolation of the information in Reference 35, we have assumed, given a two-unit melt, that the offsite dose is a factor of 1.2 times that for the single unit melt. The onsite dose received by the operators will be assumed to double for the dual melts.

11.3.2 Value-Impact Estimates for Two-Unit Melts

By making the assumptions described in the last section, estimates of the V-I ratio, net benefit, and dollars per person-rem for the two alternatives were made. This was done without revising the fault trees or re-solving the accident sequences. The results are tabulated in Table 11.2. The total "site" core melt probability was estimated to be approximately 1.24E-4 per reactor year.

Alternative 1 calls for the addition of fire retardant wrap for the B train of cabling in the cable spreading room. In addition, seismic supports would be provided for the Unit 1 condensate storage and refueling water storage tanks. These modifications would likely be repeated for Unit 2 and would double the costs.

It was assumed that two add-on dedicated residual heat removal systems would have to be installed in order to fully protect both units. The cost therefore doubles from what is estimated in Appendix J.

It should be noted that the value-impact measures presented in Table 11.2 are very rough estimates which should only be used to gauge what would happen if both St. Lucie reactors had been modeled in this study. For more precise estimates, a more

TABLE 11.2 ESTIMATE OF VALUE-IMPACT MEASURES FOR ACCIDENTS INVOLVING BOTH ST. LUCIE UNITS

Alternative No.	Reduction in Core Melt Probability	Offsite Averted Dose (p-rem)	Total Averted Dose Ratio	V-I Analysis of Offsite Costs			V-I Analysis Based on Offsite and Onsite Costs		
				V-I Ratio	Net Benefit ($ x10^{-6})	Dollars per p-rem	V-I Ratio	Net Benefit ($ x10^{-6})	Dollars per p-rem
	Δp_m	V_2	ADR_n	VIR_o	NVB_o	DPR_o	VIR_n	NBV_n	DPR_n
1	7.1E-5	251	.67	.066	- 2.0	8699	-.087	2.535	0
2	1.2E-4	397	.95	.002	- 117.0	295179	.003	-109.279	207119

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detailed modeling of the systems interactions between units would be necessary.

In summary, two general observations can be made. First, the modifications which have the highest potential for reducing the "site" risk or core melt probability for the lowest cost will be those which increase the redundancy or availability of systems shared between units. Second, while it is easy to imagine that the onsite costs such as power replacement, loss of investment, and cleanup could double for accidents involving two units, it is less clear how the avertable offsite costs might be affected. Given factors such as emergency response, land interdiction and decontamination, the overall, avertable offsite contribution to risk may not increase dramatically.

11.4 Observations from the Case Study

Although the objective of TAP A-45 is to resolve decay heat removal issues in a generic way, it is difficult to reach broadly applicable conclusions based upon one plant. However, there are a number of observations which can be made.

a) This study was generally consistent with other probabilistic risk assessments in that failures or vulnerabilities in support systems are often dominant mechanisms.

b) The fire vulnerability of the cable spreading room poses the single greatest plant vulnerability.

c) The addition of a common mode failure mechanism for each set of pumps required for decay heat removal dominates the failure of each system in the internal analysis.

d) Emergency electric power availability again appears to be a key issue, particularly in older plants where there are fewer redundant trains. This is consistent with the results found for other USIs and GSIs, e.g., A-45 and A-30.

e) The addition of emergency operating procedures for bleed and feed significantly reduced the core melt probability of St. Lucie. However, thermal-hydraulic studies conducted in support of TAP A-45 raise serious questions as to the "last ditch" nature of feed and bleed. In order for it to be effective in preventing core uncovery and melt, a decision to use it must be made quite early in the accident sequence.

f) Older plants in which separation of redundant trains often does not include compartmentalization, or where safety-related components are located outdoors, can have significant vulnerabilities. However, this case study suggests that there are some straightforward and effective remedies, although they are not necessarily inexpensive.

In this respect, as noted elsewhere in this report, the original premise of USI A-45 that there are vulnerabilities and that they can be countered is substantiated.

g) Most of the vulnerabilities identified can be reduced by modifications which involve little or no radiation exposure to construction personnel. This reflects the fact that the important vulnerabilities are in the water storage tanks and support systems which are predominantly outside containment. Of course, the independent add-on requires access to the primary system and thus installation incurs some exposure.

h) By careful planning and scheduling the sort of alternatives proposed here can be implemented during normal refueling outages, although the add-on might have to be spread across several.

i) Credit for recovery actions performed by plant personnel in response to a reactor accident had a significant effect on the results. Some of the more important recovery actions were recovering offsite power, fixing local faults of batteries or diesel generators, actuating safety systems manually, realigning auxiliary feedwater steam and feedwater flow paths, and manually opening locally failed valves.

j) A more general observation is that the methodologies of PRA and systematic analysis of special emergencies can be combined effectively in a short time scale to gain significant insights into nuclear DHR reliability. It is believed that the techniques developed here to do a simplified seismic analysis will have application beyond TAP A-45.

0193

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0197

APPENDIX A

SYSTEM DESCRIPTIONS FOR THE ST. LUCIE NUCLEAR POWER PLANT

by

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0198

TABLE OF CONTENTS

Page

1.0 INTRODUCTION A-3 200

2.0 AUXILIARY FEEDWATER SYSTEM (AFWS) A-5 202

3.0 HIGH PRESSURE INJECTION AND RECIRCULATION SYSTEM (HPIS & HPRS) A-8 205

4.0 LOW PRESSURE INJECTION AND RECIRCULATION SYSTEM (LPIS & LPRS) A-13 210

5.0 CONTAINMENT SPRAY INJECTION AND RECIRCULATION SYSTEM (CSIS & CSRS) A-14 211

6.0 PRESSURIZER RELIEF VALVES (PORV/SRV) A-17 214

7.0 CONTAINMENT FAN COOLING SYSTEM (CFCS) A-18 215

8.0 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS) A-20 217

9.0 COMPONENT COOLING WATER SYSTEM (CCWS) A-24 221

10.0 INTAKE COOLING WATER SYSTEM (ICWS) A-28 225

11.0 ELECTRIC POWER SYSTEM (EPS) A-32 229

1.0 INTRODUCTION

Each of the systems for which fault tree analyses were performed are described briefly in this appendix. A simplified schematic showing the significant flow paths and components is given when applicable with the support system dependencies identified.

The complete fault trees which were used in the analysis are available for inspection but are not reproduced here due to their size. Table A.1 gives a list of the fault trees used in the analysis and the appendix section where a system description can be found. Some main line systems have interrelationships such as the LPRS and HPRS.

Assumptions

The following general and specific assumptions were made in the construction of the event tree and fault tree models:

1) Any valves and pumps that were actuated for injection will still need actuation and power for the recirculation phase in case the injection was not required or pumps were turned off prior to the recirculation phase.

2) In order to avoid a fault tree loop where the diesel generator depends on DC power which, in turn, depends on the diesel generator, it is assumed that only component faults of the applicable DC power train result in failure of corresponding diesel generator.

3) Simultaneous shutdown of both units is not considered relative to systems shared between units. Thus all swing safeguard equipment can be used by the unit being shut down.

4) Reactor coolant pump seal failure is not considered since this is an issue which, if determined to be important, will be solved by another program. Thus neither CVCS or CCWS cooling of seals is modeled.

5) The CVCS is not needed for makeup while the pressure is being reduced for the safety injection pumps to be effective since the AFWS is used to speed up this process. The PORVs would not normally be used to get the pressure down and the pressurizer sprays are not required although they could be used.

6) The PORVs will not be used to increase the depressurization rate for secondary blowdown due to the concern that the core could be uncovered before the LPIS can inject water into the primary system.

7) The CVCS would be needed on transients eventually for makeup and for the reactor coolant pump seals but only for cold shutdown which is not covered by this analysis.

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Table A.1 System Fault Trees

Event Tree/Accident Section	Systems	Success Criteria	Sequence Symbol
2.0	AFWS	1 of 3 Pumps	L
3.0	HPIS	1 of 3 Pumps	D_1
	HPRS	1 of 3 Pumps + 1 of 2 HX	H_1
	HPRS-No HX	1 of 3 Pumps	H_1'
	FNB	1 of 3 Pumps	E
4.0	LPIS	1 of 2 Pumps	D_2
	LPRS	1 of 2 Pumps + 1 of 2 HX	H_2
	LPRS-No HX	1 of 2 Pumps	H_2'
5.0	CSIS	1 of 2 Pumps	C_1
	CSRS	1 of 2 Pumps + 1 of 2 HX	C_2
	CSRS-No HX	1 of 2 Pumps	C_2'
6.0	PORV	2 of 2	P_1
7.0	CFCS	1 of 4 Fans	Y_1
8.0	ESFAS		
9.0	CCWS	1 of 3 Pumps + 1 of 2 HX	
10.0	ICWS	1 of 3 Pumps	
11.0	EPS		

8) In the event L defined as the AFWS and secondary steam relief, the secondary steam relief is not fault tree modeled since there is adequate redundancy, i.e., ADVs, or safety relief valves, or the steam dump to the condenser given condenser vacuum is available except in the case of a double steam generator tube rupture which is not being addressed.

9) Containment failure does not imply core melt due to containment sump flashing as has been assumed in many past PRAs. This permits the containment systems to be shown in a separate event tree.

10) The AFWS is sufficient to remove all decay heat without the RHR heat exchangers during the shutdown heat removal phase, i.e., to hot shutdown.

2.0 AUXILIARY FEEDWATER SYSTEM (AFWS)

The auxiliary feedwater system is designed to:

1) Provide feedwater for the removal of sensible and decay heat from the reactor coolant system (during normal or natural circulation system cooldown

2) Provide sufficient feedwater capacity to permit plant cooldown to 325°F, assuming a single active failure and loss of offsite power

3) Withstand design basis earthquake loads without loss of function.

The major active components of the system consist of one greater than full flow capacity and two full flow capacity auxiliary feedwater pumps. The larger pump is driven by a noncondensing steam turbine. The turbine receives steam from the main steam lines upstream of the isolation valves and exhausts to atmosphere. The two motor driven pumps are powered from the emergency generators in case of a loss of normal power. The pumps take suction from the condensate storage tank and discharge to the steam generators. Refer to Figure A.1.

The turbine-driven pump is capable of supplying auxiliary feedwater flow to the steam generators for the total expected range of steam generator pressure (985 psig to 50 psig) by means of a variable speed turbine driver controlled by a variable speed hydraulic governor. The turbine operates through a speed range of 3600 to 2000 rpm when supplied with saturated steam from 985 to 50 psig respectively. The turbine-driven pump relies solely on a DC power supply; the valves associated with the turbine-driven pump also are powered from a DC source. Each motor-driven pump generally supplies feedwater to one steam generator. A cross connection has been

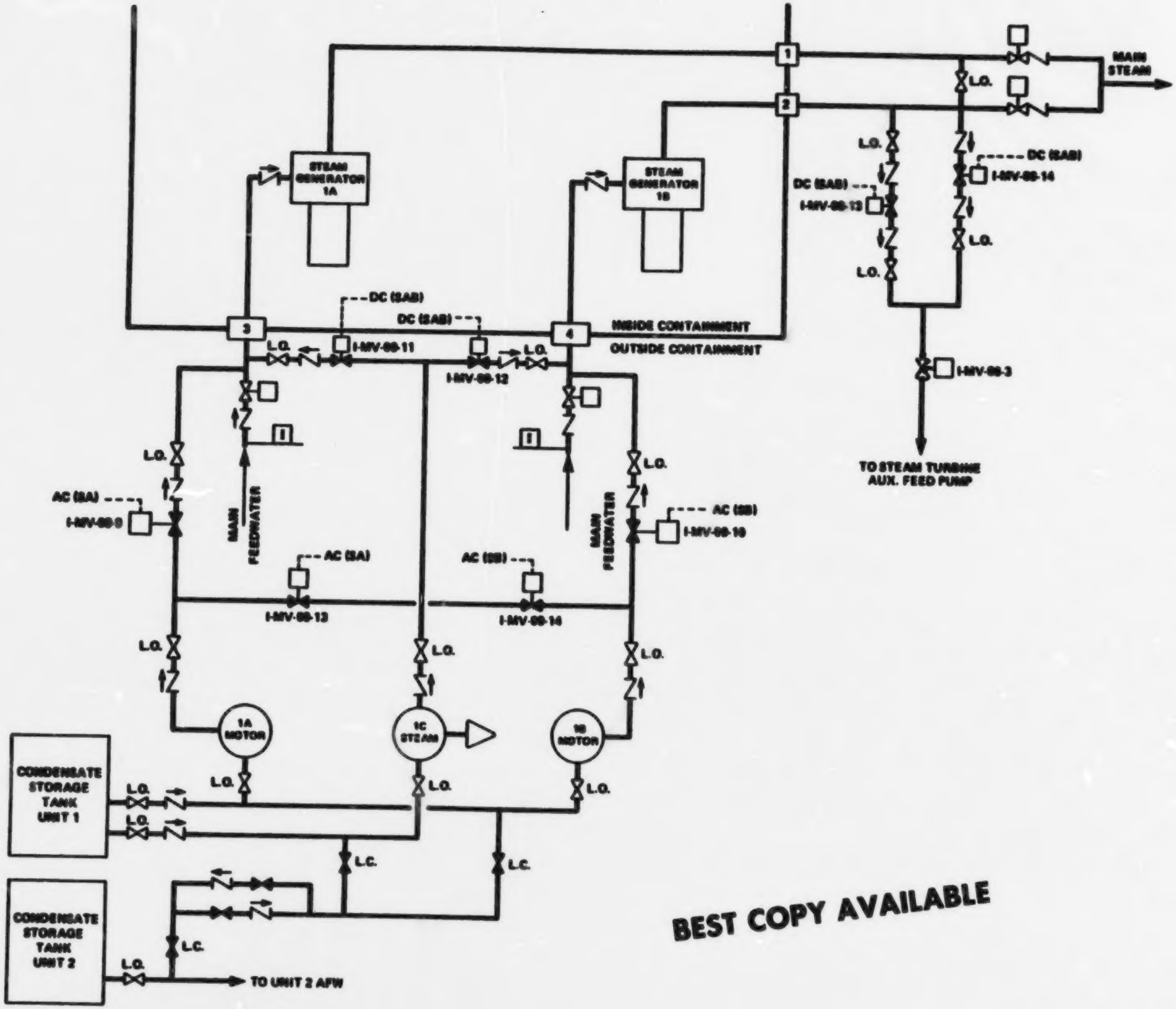
MAIN STEAM
1
2
L.O.
DC (SAB)
I-MV-08-14
DC (SAB)
I-MV-08-13
L.O.
L.O.
L.O.
I-MV-08-3
TO STEAM TURBINE AUX. FEED PUMP
STEAM GENERATOR 1A
STEAM GENERATOR 1B
3
4
INSIDE CONTAINMENT
OUTSIDE CONTAINMENT
DC (SAB)
DC (SAB)
L.O.
I-MV-09-11
I-MV-09-12
L.O.
MAIN FEEDWATER
MAIN FEEDWATER
L.O.
L.O.
AC (SA)
I-MV-09-9
AC (SB)
I-MV-09-10
AC (SA)
AC (SB)
I-MV-09-13
I-MV-09-14
L.O.
L.O.
L.O.
1A MOTOR
1C STEAM
1B MOTOR
L.O.
L.O.
L.O.
CONDENSATE STORAGE TANK UNIT 1
L.O.
L.O.
L.C.
L.C.
CONDENSATE STORAGE TANK UNIT 2
L.C.
L.O.
TO UNIT 2 AFW

Figure A.1 Auxiliary Feedwater System Schematic

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provided to enable the routing of the flow of the two motor-driven pumps to one steam generator. The turbine-driven pump supplies feedwater to both steam generators by means of two separate lines each with its own control valve. The control of auxiliary feedwater flow and steam generator level is normally accomplished by means of control-room operated control valves each sized to pass the full flow. A local control system is provided to facilitate shutdown if the control room is not accessible.

During normal operation, feedwater is supplied to the steam generators by the main feedwater system. If this system is unavailable due to loss of feedwater pumps or offsite power, an alarm is sounded in the control room, and the automatic sequence of the AFW auto-start circuit will initiate on low steam generator level. If the auxiliary feedwater pumps were operating when offsite power was lost, the pumps would automatically restart using diesel generator power.

The steam generated during decay heat removal and cooldown after a loss of offsite power will be discharged through the atmospheric dump valves, except for that steam used by the auxiliary feed pump. If offsite power and the main condenser are available, the condenser may be used as a heat sink.

The auxiliary feedwater system is designed to provide feedwater for the removal of sensible and decay heat from the reactor coolant system and the system can cool the reactor coolant system to 325°F. The auxiliary feedwater system may also be used for normal system cooldown to 325°F. Reactor decay heat and sensible heat are transferred to the steam generators by natural circulation of the reactor coolant if power is not available for the reactor coolant circulating pumps.

The storage capacity of the condensate storage tank is 250,000 gallons, a sufficient quantity to meet the requirements for decay heat removal and cooldown of the nuclear steam supply system. After a loss of normal feedwater, approximately 110,000 gallons of water are required to permit cooldown to 325°F following a reactor trip. A crosstie is provided between the Unit 1 auxiliary feedwater pumps and the Unit 2 condensate storage tank to add additional system flexibility.

Analysis indicates that 138,000 gallons are used for 8 hours of hot standby following a reactor trip. This includes 28,000 gallons to raise the water level to the normal operating level from the low level it would be at because of the time lapse in initiation of the auxiliary feedwater system following loss of main feed pumps.

The auxiliary feedwater system is designed as seismic Class I and is capable of withstanding tornado wind loading. The auxiliary feedwater pumps are protected from design basis missiles by protective shielding. The auxiliary feedwater system is protected from flooding by placing all system components susceptible to flooding damage above the probable maximum flood level or by providing flood protection barriers around them.

3.0 HIGH PRESSURE INJECTION AND RECIRCULATION SYSTEM (HPIS and HPRS)

The high pressure safety injection pumps function to inject borated water into the reactor coolant system if a break occurs in the reactor coolant pressure boundary. For small breaks, the reactor coolant system pressure remains high for some period of time following the leak, and the high pressure safety injection pumps will ensure that the injected flow is sufficient to meet the design criteria. The high pressure safety injection pumps are also used during recirculation to maintain water level over the core following the LOCA.

The high pressure safety injection pumps are sized such that one pump will deliver saturated water at a flow rate sufficient to maintain the core flooded, matching decay heat boil-off at the time the safety injection system switches to the recirculation mode not less than 20 minutes after LOCA.

The pumps have mechanical seals to minimize shaft leakage and are provided with leakoffs to contain the leakage which is drained to the waste management system. To ensure extended seal life, a portion of the pump discharge flow is cooled by component cooling water and is recirculated to the seals. The pump motor is capable of starting and accelerating the pump to full speed in 8 seconds with only 75 percent of rated voltage. The pumps are provided with minimum flow protection to prevent damage resulting from operation against a closed discharge.

The design temperature for the high pressure safety injection pumps is based upon the saturation temperature of the reactor coolant at the containment design pressure, about 300°F, plus a design tolerance of 50°F, resulting in design temperature of 350°F. The pumps are specified to accept the 10 second temperature transient produced by the transfer of the recirculation mode when the pump suction is switched from the refueling water tank to the containment sump. The design pressure for the high pressure pumps is based upon the sum of the containment spray pump shutoff head and the shutoff head of the high pressure pump.

The pumps have a design pressure of 1600 psig and a design flow rate of 345 gpm each with a design head of 2500 ft. The maximum flow rate is 640 gpm with a head of 1300 ft.

The available NPSH for the high pressure safety injection pumps, operating at the system runout flow of 640 gpm, is approximately

24 feet. The NPSH required, as determined by the pump test curves furnished by the manufacturer, is 20 feet at 640 gpm.

Operation of the safety injection system is fully automatic on receipt of a safety injection actuation signal, requiring no action by the operator other than to reaffirm that automatic action of equipment and the notification and protection of plant personnel is underway.

In order to limit the effects of a passive failure of either the piping or valves in the system, the safety injection and containment spray equipment is divided and located in two watertight rooms. The safety injection pumps, heat exchangers, and piping are arranged in these compartments so that flooding will not result in loss of minimum required safety features.

The safety injection flow path to the reactor vessel is through a safety injection nozzle on each of the four reactor coolant system cold leg pipes as seen in Figure A.2.

The components in the safety injection system have no service function during power operation. Therefore, no change in the operating mode of any part of the system is required for the system to perform its intended function.

All valves that are not required to operate on initiation of safety injection or recirculation in the safety injection flow path are locked in the safety injection position during operation. Administrative controls ensure that the valves are locked in the correct position.

Two redundant high pressure safety injection trains are provided. Pump 1B and the associated four injection valves (HCV-3616, -3626, -3636, and -3646) operate from diesel generator 1B. Pump 1A and the associated injection valves (HCV-3617, -3627, -3637, and -3647) operate from diesel generator 1A.

This ensures the automatic operation of one complete train in the unlikely event of simultaneous loss of offsite power and failure of any active component, including failure of one emergency diesel generator. The third high pressure pump is an installed spare and will be permanently aligned to HPSI header "B". Interlocks will be provided to prevent the starting of the third pump when 4160 volt AB Bus is supplied from 4160 volt Bus A. The header cross-over motor operated valve V-3655 will be locked closed, and its open position will be annunciated in the Control Room.

The major safety injection system components located inside the containment building are the four safety injection tanks. The safety injection tanks are located outside the secondary shield wall at elevation 48 ft. The high and low pressure safety

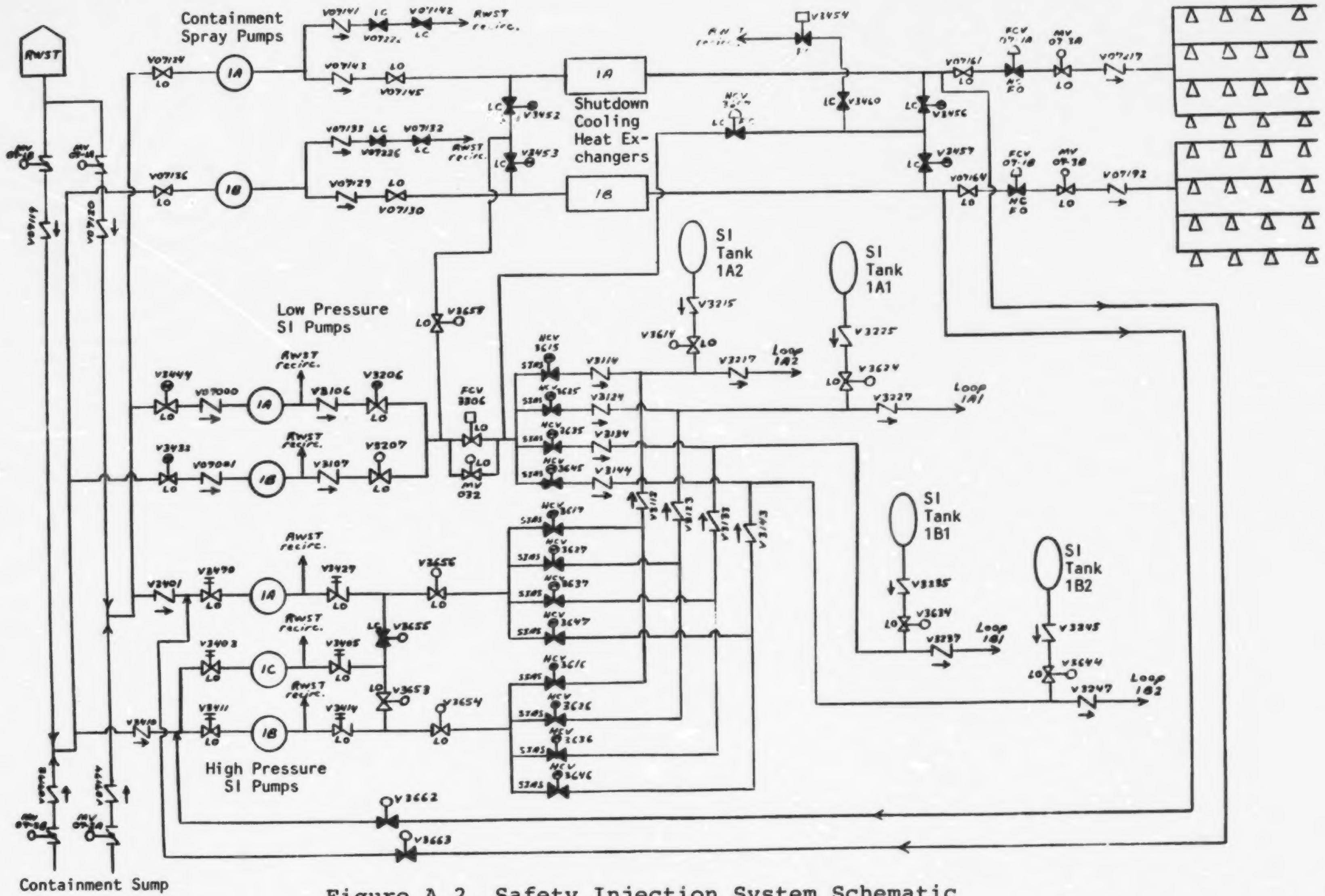
RWST
Containment Spray Pumps
V07124
LO
1A
V07141
LC
V07142
LC
V07226
RWST recirc.
V07143
LO
V07145
V07136
1B
V07133
LC
V07132
V07226
RWST recirc.
V07129
LO
V07130
LC
V3452
V3453
1A
1B
Shutdown Cooling Heat Exchangers
V3454
HCV
LC
V3460
V3456
V3457
V07161
FCV 07-1A
NC FO
MV 07-3A
LO
V07217
V07164
FCV 07-1B
MV 07-3B
V07192
MV 07-1B
MV 07-1A
V07119
V07120
SI Tank 1A2
V3215
V3614
V3217
Loop 1A2
SI Tank 1A1
V3225
V3624
V3227
Loop 1A1
Low Pressure SI Pumps
V3658
V3444
V07000
RWST recirc.
V3106
V3206
V3432
V07001
V3107
V3207
FCV 3306
MV 032
HCV 3615
SIAS
V3114
HCV 3625
V3124
HCV 3635
V3134
HCV 3645
V3144
V3113
V3123
V3133
V3143
HCV 3617
HCV 3627
HCV 3637
HCV 3647
SI Tank 1B1
V3235
V3634
V3237
Loop 1B1
SI Tank 1B2
V3245
V3644
V3247
Loop 1B2
V3401
V3470
V3427
V3656
V3655
V3403
1C
V3405
V3653
V3410
V3411
V3414
V3654
HCV 3616
HCV 3626
HCV 3636
HCV 3646
High Pressure SI Pumps
V3662
V3663
MV 07-2B
MV 07-2A
Containment Sump

Figure A.2 Safety Injection System Schematic

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injection pumps are located at elevation -10 ft in the reactor auxiliary building. This location maximizes the available NPSH for the safety injection pumps. It also permits sloping the containment sump suction pipe lines down to minimize activity buildup due to fission product deposition in the lines.

Safety injection is automatically initiated upon a safety injection actuation signal (SIAS) which is produced by coincidence of two-of-four low pressurizer pressure signals or two-of-four high containment pressure signals. Low pressurizer pressure and high containment pressure are indicative of a LOCA. The reactor is tripped by the reactor protective system and the turbine is tripped coincident with a reactor trip.

For a plant shutdown, the SIAS is blocked by the operator after cooldown and partial depressurization. The block is automatically removed when system pressure increases above the specified pressure.

With the plant in a shutdown/cooled down condition, loss of coolant would not produce a SIAS. However, the operator would be alerted by a combination of the following alarms and/or indicators:

a) low pressurizer level alarm
b) low reactor coolant system pressure
c) high containment pressure alarm
d) containment high radiation alarm
e) containment sump level alarm
f) shutdown cooling temperature
g) temperature of component cooling water to and from containment cooling system coolers
h) low volume control tank level alarm

For large breaks in the reactor coolant system, all of the above alarms and indications will be present. As break size decreases, so will the number of indicators and alarms. As a minimum, the indicators and alarms that will be actuated in response to a break equal to or less than the capacity of one charging pump increase in containment activity, increase in sump water level, increase in temperature of the component cooling water to and from the containment cooling system, and decrease in volume control tank level.

On being alerted to a LOCA, the operator would manually initiate the SIAS, either by removing the block on low-low pressure or by means of the manual initiation switch provided for each channel.

The safety injection system transfers automatically from the injection mode of operation to the recirculation mode of operation on receipt of the recirculation actuation signal (RAS). The RAS is produced by a low level signal from the refueling water tank.

0208

The recirculation mode of operation must not be initiated until at least 20 minutes have elapsed since initiation of safety injections. At this time the core decay heat will be less than the heat removal capacity of one high pressure safety injection pump taking suction from the containment sump. For this reason the injection mode of operation must be maintained for at least 20 minutes and sufficient refueling water tank volume for this operation must be provided.

On receipt of the RAS, the containment sump isolation valves are opened and the refueling water tank outlet valves are closed. Opening the sump isolation valves results in a transfer of the safety injection and spray pump suction from the tank to the containment sump. Closing the outlet valves provides double isolation between the recirculating coolant and the tank.

The RAS also acts to shut down the low pressure safety injection pumps and closes the isolation valves in the pump recirculation line (V-3659 and V-3660). Recirculation flow is normally provided by at least one high pressure safety injection pump. At the discretion of the operator a portion of the cooled water from the containment spray system may be directed to the high pressure pump suction. This is not necessary to meet core cooling requirements but is a preferred mode of operation.

St. Lucie Unit 1 currently has procedures for a mode of operation known as Bleed and Feed. In this scenario, all secondary side cooling to the steam generators is lost following a transient. In Bleed and Feed, core cooling is accomplished by bleeding the primary coolant out of the pressurizer through the PORVs which reduces the RCS pressure below the HPSI pump shutoff head. This then allows coolant to be fed into the RCS by the HPSI system. This process is repeated as the RCS pressure rises and falls. The procedures for this mode of operation at St. Lucie Unit 1 utilize the small LOCA HPI success criteria of one out of three high head pumps. Containment heat removal is achieved by one of two spray pumps or one of four emergency fan coolers.

The various modes of HPSI system operation correspond to headings on the event trees used in determining the accident sequences that were analyzed in this study. These headings are: D_1, H_1, H_2' and E which represent HPIS, HPRS with secondary side cooling of the shutdown cooling heat exchanger (HX), HPR without HX cooling and Bleed and Feed, respectively. For HPI, the success criteria requires the operation of one of the three HPSI pumps injecting into one RCS cold leg. In the cases where Bleed and Feed is considered, the additional requirement is that both of the PORVs be available for bleeding the RCS.

0209

A major assumption incorporated into these models concerns the recirculation phases of ECC, which are HPRS and LPRS modes of operation. The assumption was made that containment failure does not result in the loss of enough water in the containment sump to fail the RHR and/or SI pumps due to cavitation. The rationale for this is that although some water in the sump may boil away on containment failure, enough would remain such that recirculation cooling is unaffected.

4.0 LOW PRESSURE INJECTION AND RECIRCULATION SYSTEM (LPIS and LPRS)

The low pressure safety injection pumps serve two functions. The first is that of injecting large quantities of borated water into the reactor coolant system during an emergency involving a large pipe rupture. Sufficient flow is delivered under these conditions to satisfy functional requirements. The second function of the low pressure safety injection pumps is to provide shutdown cooling flow through the reactor core and shutdown cooling heat exchangers.

The low pressure safety injection pumps are sized to provide adequate shutdown cooling flow. The flow available with one of the two low pressure safety injection pumps is sufficient to maintain a core ΔT of less than the full power ΔT of 55.4°F at the initiation of shutdown cooling 3.5 hours after shutdown. Maintenance of the core ΔT at less than full power ΔT results in a lesser demand on the structural design of the core than design operations.

The design temperature for the low pressure safety injection pumps is based on the temperature of the reactor coolant at the initiation of shutdown cooling, about 300°F nominal, plus 50°F. The design pressure for the low pressure pumps is based upon the sum of the maximum pump suction pressure, which occurs at the initiation of shutdown cooling, plus the pump shutoff head.

The low pressure safety injection pumps are horizontal, single-stage centrifugal units with mechanical face seals backed up by a bushing, with a leakoff to collect the leakage past the seal. To ensure extended seal life, a portion of the pump discharge is cooled by component cooling water and is recirculated to the seals.

The pump motor is capable of starting and accelerating the pump to full speed within 8 seconds with 75% of rated voltage. The pumps are provided with drain and flushing connections to permit reduction of radiation levels before maintenance. The pressure containing parts are fabricated from stainless steel; the internals are selected for compatibility with boric acid solutions. The pumps are provided with miniflow protection to prevent damage when starting against a closed system.

0210

Each pump has a design pressure of 500 psig and a design flow rate of 3000 gpm with 350 ft head. The maximum flow rate is 4500 gpm with 235 ft head.

Active components necessary for low pressure safety injection are sized and arranged to provide two trains capable of providing the required flow. Two motor operated injection valves and one low pressure safety injection pump operate from each diesel generator (Pump 1A, valves HCV-3615 and HCV-3625; pump 1B, valves HCV-3635 and HCV-3645). Low pressure safety injection flow will therefore be available assuming the failure of one diesel generator occurring simultaneously with the loss of off-site power. Passive components such as piping do not require redundancy since the low pressure safety injection system is not required to operate during the recirculation mode of core cooling. Header valve FCV-3306 is a fail-open pneumatically operated valve and the air signal to close is locked out electrically by a key switch in the control room. At the time of the construction permit, a single valve that is locked out in accordance with administrative control procedures was considered a passive component, thus not subject to redundance requirements which accommodate single active component failure.

The Staff has required in their safety evaluation report that a redundant valve be provided. A motor operated locked open 10 inch bypass valve is in place for this purpose. The addition of this bypass valve obviates the need for administrative control of FCV-3306.

The addition of the 10 inch motor operated bypass valve was in concurrence with the requirements of 10CFR50.109. Safety injection pumped flow is not available until 30 seconds following a safety injection actuation signal (SIAS). This allows for the startup and loading of diesel generators in the event of a loss of offsite power. The diesel generators and emergency power system are designed to start the safety injection pumps within 13 seconds after SIAS with loss of offsite power. The HPSI valves have an opening time not greater than 10 seconds. The LPSI valves have an opening time not greater than 15 seconds.

5.0 CONTAINMENT SPRAY INJECTION AND RECIRCULATION SYSTEM (CSIS and CSRS)

The containment heat removal system consists of the containment spray system and the containment cooling system and is designed to:

a) prevent the containment pressure from exceeding its design value following a LOCA, assuming a single active or passive failure

b) withstand the design basis earthquake loads without loss of function

0211

c) withstand post-accident environmental conditions without loss of function

d) permit inspection and testing at appropriate intervals to assure system integrity and functional capability.

The containment spray system consists of two independent and redundant subsystems. The heat removal capacity of either of the two subsystems is adequate to keep the containment pressure and temperature below design values and to bring the containment pressure below 10 psig within 24 hours after any size break in the reactor coolant system piping up to and including a double-ended break of the largest reactor pipe, assuming unobstructed discharge from both ends.

The containment cooling system consists of four fan-coil cooling units, a ducted air distribution system, and the associated instrumentation and controls. The heat removal capacity of the containment cooling system is adequate to keep the containment pressure and temperature below design values and to bring the contaiment pressure below 10 psig within 24 hours after any size LOCA. This system is described in Section 7.0 of the Appendix.

The containment spray system consists of two independent and redundant subsystems each containing a spray pump, shutdown heat exchanger, piping, valves and spray header as shown in Figure A.10. The system has two modes of operation which are:

a) the initial injection mode, during which the system sprays borated water from the refueling water storage tank into the containment.

b) the recirculation mode, which is automatically initiated by the recirculation actuation signal (RAS) after low level is reached in the refueling water tank. During this mode of operation, suction for the spray pumps is from the containment sump.

Containment spray is automatically initiated by the containment spray actuation signal (CSAS) which is a coincidence of the safety injection actuation signal (SIAS) and the high-high containment pressure signal.

The CSAS is composed of two actuation channels (A and B) with each channel starting its associated spray pump (A or B) and opening its associated spray isolation valve. If required, the operator can manually actuate the system from the control room.

The heat removal capacity of either of the two subsystems is adequate to keep the containment pressure and temperature below design values. Each of the two spray pumps is designed to deliver 2700 gpm of borated cooling water (approximately 1720 ppm boron) to the containment atmosphere against a total dynamic head (TDH) of 470 ft. The heat removal capacity is approximately 215 x 10^6 Btu/hr for a peak containment temperature of 259°F and an assumed incoming spray temperature (with water drawn from the refueling water tank) of 100°F; the heat removal capacity is approximately 240 x 20^6 Btu/hr for an assumed incoming spray temperature of 85°F.

The containment spray pumps initially take suction from the refueling water tank. When low level is reached in the refueling water tank, sufficient water has been transferred to the containment, at least 305,600 gal, to allow for the recirculation mode of operation. Spray pump suction is automatically realigned to the containment sump upon RAS. Automatic realignment of suction requires opening the valves in the rcirculation lines and closing the valves at the outlets of the refueling water tank. To assure adequate supply of water for the pumps during suction transfer, the sump valves are designed to be fully open within 30 seconds and the tank valves to be fully closed at 90 seconds following RAS. This is accomplished by using different gear ratios on the motor operators.

During the recircuation mode, the spray water is cooled by the shutdown heat exchangers prior to discharge into the containment. The shutdown heat exchangers are cooled by the component cooling system. The basis for sizing the shutdown heat exchangers is the refueling mode of operation since the heat exchanger tube to shell side ΔT is lowest in that mode. The heat removal capacity of the heat exchangers is more than adequate for post-accident conditions.

The spray water is discharged into the containment dome through spray nozzles arranged on headers. The spray nozzles are of the open throat design and are not subject to clogging. The containment spray system operation is based on the spray water being raised to the temperature of the containment air-stream mixture. This occurs when the spray water falls through the steam-air atmosphere within the containment. The spray nozzles in each of the two headers are designed to deliver droplets of approximately 700 microns (3.93 x 10^{-5} in.) mean diameter with the spray system operating at design conditions and the containment at design pressure. In order that the spray droplets attain thermal equilibrium with the steam-air mixture during the fall, approximately 70 ft has been provided between the spray nozzles and the top of the steam generators.

The spray water is directed toward the containment sump through a number of flow paths. Any water entering the refueling cavity leaves the cavity through sleeves in the side walls. Any water falling within the secondary shield wall flows into a trench located outside the secondary shield through openings in the shield wall. The trench is sloped toward the containment

0213

sump. The minimum water level elevation expected in the containment following a LOCA is 25.3 ft. The water level could reach elevation 27.3 ft if the entire useable volume of the refueling water tank is sprayed and injected into the containment.

All portions of the spray system which are designed to recirculate radioactive water collected in the containment sump are designed to operate in the radiation environment associated with the maximum hypothetical accident (MHA).

All components of the containment heat removal system have been designed as seismic Class I and are installed in seismic Class I structures, Seismic Class I has been specified in purchase specifications, and vendors have substantiated either through test, calculational and/or operational data that system components will remain operable under the design basis earthquake loads.

6.0 PRESSURIZER RELIEF VALVES (PORV/SRV)

The reactor coolant system has three safety valves to provide overpressure protection. These valves which are flange mounted on the top of the pressurizer, are totally enclosed, back pressure compensated spring loaded safety valves meeting ASME Code requirements.

The safety valves pass sufficient pressurizer steam to limit the reactor coolant system pressure to 110% of design 2750 psig following a complete loss of turbine generator load without simultaneous reactor trip. A delayed reactor trip is assumed on a high pressurizer pressure signal. To determine maximum steam flow through the pressurizer safety valves, the main steam safety valves are assumed to be operational.

The valves have a set point pressure of 2485 psig and a flow capacity of 200,000 lb/hr at the set pressure.

There are two power operated relief valves in the reactor coolant system. The valves are solenoid-operated power relief valves. The two half-capacity valves are located in parallel pipes which are connected to the pressurizer relief valve nozzle on the inlet side and to the relief line piping to the quench tank on the outlet side. A motor actuated isolation valve is provided upstream of each of the relief valves to permit isolating the valve for maintenance or in case of valve failure.

The PORVs have a design pressure of 2485 psig and a set pressure of 2385 psig with a capacity of 153,000 lb/hr.

To meet the requirements for maintaining operability of these valves following loss of offsite power, the power operated

0214

relief valves (V-1402, V-1404) are supplied from Class IE DC power and the motor actuated isolation valves (V-1403, V-1405) are supplied by Class IE AC power.

The two power operated relief valves relieve sufficient pressurizer steam during abnormal transients to prevent opening of the reactor coolant system safety valves. The relief valves are actuated by the high reactor coolant system pressure trip signal.

The capacity of the power operated relief valves has been selected to pass the maximum steam surge associated with a continuous CEA withdrawal starting from low power. Assuming that a reactor trip is effected on a high pressure signal, the capacity of the power operated relief valves prevents the opening of pressurizer safety valves. The total relief valve capacity is also large enough that the safety valves do not open during a loss of load from full power. This assumes normal operation of the pressurizer spray system, and reactor trip on high pressurizer pressure.

The PORVs appear in the event tree analysis both alone and in conjunction with the HPSI system (as part of the analysis of the bleed and feed mode of operation) with the success criteria for the PORVs assumed to be a two out of two requirement in both cases. The availability of the PORVs becomes important only when all secondary side cooling for the steam generators has been lost. The PORVs are then relied upon to keep the RCS pressure low enough (in either a transient or small LOCA) such that the HPSI system is capable of injecting water for ECC. Any time the PORVs are called upon for aiding HPSI, it must be performed manually by the operators in the control room. There is no automatic actuation system associated with the PORVs except for their pressure setpoint.

There are two events listed on the St. Lucie event trees which refer to the SRVs. These events are not modeled with fault trees, as are the PORVs, but are calculated directly as a failure probability. These events appear under the headings of P and Q, which are: SRVs fail to open and SRVs fail to reclose, respectively. These events were attributed to the SRVs rather than the PORVs since in this analysis the PORV block valves are assumed to be normally closed at St. Lucie.

7.0 CONTAINMENT FAN COOLING SYSTEM (CFCS)

The system consists of four fan coolers, a ducted air distribution system and the associated instrumentation and controls. Each fan cooler consists of two banks of cooling coils, casing, centrifugal fan and motor.

The containment cooling system is required to function as a separate and independent means of containment heat removal during a LOCA. The system represents a diversity in design from the containment spray system. With four fan cooling units operating, the containment cooling system is capable of removing heat from the containment atmosphere at a rate of approximately 240 x 10^6 Btu/hr under post-accident conditions.

Each cooling coil bank is made up of coil sections connected to supply and return manifolds of the component cooling water system. The flow to the coils from the component cooling water system is monitored by a low flow indicator. Coils are horizontal tube, vertical plate-fin type, and are mounted on a structural frame. Cooling coils are provided with individual drain pans and drain piping to prevent flooding of lower coils by condensed water cascading from upper coils. Condensation is drained to the reactor cavity sump. The cooling coils have been designed to withstand a pressure of 400 psig.

The casing of each containment fan cooler is reinforced to withstand a pressure differential in any direction of 2 psi. The fans are centrifugal type, direct-driven, with backwardly curved airfoil blades to provide a non-overloading characteristic. Fan housing is of welded steel construction and reinforced to withstand the transient pressure rise following a LOCA. Fan motors are totally enclosed fan-cooled type with integrally mounted water-to-air heat exchanger to form an entirely enclosed cooling system. Cooling water to the fan motors is supplied from the same component cooling water line supplying the fan-coil unit.

The duct distribution system is arranged to promote mixing of the containment air and includes a common ring header to assure continuity of design air flows at all outlets. The adequacy of the duct distribution system to provide mixing of the containment air is confirmed by pre-operational testing. Ducts are of welded construction, reinforced and provided with pressure relief dampers to withstand LOCA induced pressure transients.

The containment ring header is designed to attenuate high pressure transmission from the steam generator area through the duct by having blowout panels in the ductwork from the header to the steam generator and cavity cooling system, and by providing gravity dampers rated at 2 psi differential at the point of juncture between the ring header and the aforementioned ducts.

The fan coolers are located outside the secondary shield wall in different quadrants of the containment: three on floor elevation 45 ft, and one on the operating floor at elevation 62 ft. This arrangement provides separation and minimizes recirculation between units. During normal conditions, any three of the four containment fan coolers are in operation. Each unit is sized to remove one-third of the total normal heat load or one-fourth of a large LOCA accident load. It is estimated that

0216

one fan could remove the heat loads produced by a small LOCA or transient scenario. Containment ambient temperature is limited to an average of 120°F when the units are supplied with component cooling water at 100°F. The fourth fan cooler is automatically started upon receipt of a safety injection actuation signal.

The PSAR originally stated that the containment cooling system would be actuated on containment spray actuation signal (CSAS). Upon review, it was decided that since the containment spray and containment cooling systems are provided for diversity of containment heat removal, diverse actuating signals (CSAS and SIAS) are apropriate.

Independent supply and return component cooling lines are provided for each cooler from two headers outside the containment. Butterfly valves in each line permit individual isolation. Cooling water flow is not modulated.

8.0 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS)

The engineered safety features actuation system (ESFAS) monitors selected plant parameters and provides actuation signals to the actuated components in the ESF system when the plant parameters reach preselected setpoints.

The engineered safety features actuation systems (ESFAS) consist of devices and circuitry needed to actuate the following signals:

a) Safety Injection Actuation Signal (SIAS)
b) Recirculation Actuation Signal (RAS)
c) Containment Spray Actuation Signal (CSAS)
d) Containment Isolation Signal (CIS)
e) Main Steam Isolation Signal (MSIS)

Each of the engineered safety features actuation systems consist of four measurement channels (designated MA, MB, MC and MD) for each input parameter, two logic matric systems (SA and SB) and two actuation channels (A and B).

Each measurement channel consists of a sensor, power supply and bistable unit arranged in a current flow loop circuit. The bistable unit provides a digital signal to logic matrices where signals from all four measurement channels for that parameter are combined in a two-out-of-four logic network. Isolation devices are provided to maintain separation between the measurement channels and the logic matrices. Logic matrices (SA or SB) provide initiation signals to their associated actuation channels (A or B) when the logic for the particular signal is satisfied.

The safety injection actuation signal (SIAS) logic has four independent pressurizer pressure transmitters (PT-1102A, B, C,

D) and four independent containment pressure transmitters (PT-07-2A, B, C, D) to provide signal inputs. Safety injection is initiated either by two-out-of-four low-low pressurizer signals (1585 psig) or two-out-of-four high containment pressure signals (5 psig).

Safety injection is actuated by two independent systems of diverse principle (low-low pressurizer pressure and high containment pressure). Manual actuation may be initiated from a single combination control switch and pushbutton for each actuation channel or by an individual pump and valve control switches located in the control room. In addition, a SIAS initiates a Containment Isolation Signal (CIS).

In the event of a LOCA, the SIAS automatically:

a) starts two high pressure pumps.

b) opens four high pressure, four auxiliary high pressure and four low pressure header valves. The valve control circuit permits manual adjustment of the safety injection water flow to satisfy emergency core cooling requirements.

c) closes four safety injection tank check valve leakoff valves and open four safety injection tank isolation valves if not fully open.

d) closes FW isolation valves and FW pump discharge valve.

e) starts emergency diesel generators and actuates diesel generator load shedding.

f) starts three charging pumps, two boric acid pumps and actuates associated valves. The automatic pump trip circuits are inhibited by SIAS.

g) actuates component cooling, intake cooling water pumps and associated valves.

h) actuates emergency core cooling system (ECCS) pump area ventilation system.

i) actuates containment cooling fan.

j) actuates miscellaneous equipment.

k) initiates the Containment Isolation Signal (CIS).

Separate control switches are provided for manual actuation of each valve and pump in the control room.

The coincidence of two low water level signals from any of four independent level sensing devices (LT 07-2A, 2B, 2C, 2D) on the

0218

refueling water tank initiates the recirculation actuation signal (RAS). The refueling water tank level switch (LIS-07-1) provides sufficient diverse information to enable manual initiation of RAS. RAS automatically transfers the suction of the safety injection and containment spray pumps to the containment sump by opening the two sump outlet valves, closing the refueling water tank outlet valves, and closing the pump miniflow recirculation valves to the refueling water tank. Concurrent with transfer of pump suction from the refueling water tank to the contaiment sump, the low pressure safety injection (LPSI) pumps are automatically stopped on RAS. The RAS measurement channels and logics are designed to "energize and actuate" due to the special requirement of the actuated components. The refueling water tank valves (I-MV-07-1A, 1B) and containment sump suction valves (I-MV-07-2A, 2B) are required to change position to accomplish transfer to the recirculation mode of post-accident cooling. It is essential that the refueling water tank valves remain open, the sump valves remain closed, and the safety injection pumps continue operating during the injection phase to ensure that the required quantity of refueling water is injected into the containment sump prior to recirculation. By designing the RAS as "energize to actuate," a loss of power on one 125 volt DC bus will not cause spurious RAS initiation which could possibly interrupt cooling to the core and containment before adequate water is available in the sump for recirculation. Valve circuitry permits closing any containment sump suction line or RWT outlet line after an RAS from either the control room or a local control station. Control room annunciation is provided to alarm failure of the sump suction valve to open or of the RWT outlet valve to close after an RAS; valve overload; power failure; opening of a sump valve or closure of a RWT outlet valve without receipt of an RAS; or, closure of any header after an RAS.

As the system is designed, in the event of loss of power on one 125 volt DC bus, the RAS logic and actuation channel which is not associated with the faulted DC bus is initiated at the proper time by the two measurement channels served by the unaffected bus. No single failure can prevent initiation of both RAS channels.

The sequencing between the refueling water tank valves (I-MV-07-1A, 1B) and the containment sump valves (I-MV-07-2A, 2B) is provided to ensure sufficient NPSH to the safety injection pumps during transfer from refueling tank water supply to containment sump. The containment sump valves (I-MV-07-2A, 2B) have motor operators that open the valves in 30 seconds and the rrefueling water tank outlet valves (I-MV-07-1A, 1B) have motor operators that close the valves in 90 seconds.

Each RAS actuation channel can also be initiated manually from the control room.

0219

The containment spray actuation signal (CSAS) is initiated by a coincidence of containment high-high pressure (10 psig) and SIAS. Containment atmosphere pressure is monitored by four independent measurement channels. Measurement channel signals are combined in two-out-of-four logic matrices. The output signals from logic matrices (A or B) are combined with the corresponding signal from the two SIAS actuation channels (A or B) to provide two independent actuation channels for CSAS. Each CSAS channel (A or B) initiates operation of the associated spray pump (A or B) and the associated valving. Each spray system isolation valve (FCV-07-1A and 1B) is opened by its associated CSAS actuation channel (A or B).

The CSAS containment pressure measurement channels and CSAS logics are designed as "energize to actuate" to prevent spurious spray system operation on loss of power to one 125 volt DC bus.

The 125 volt DC system is designed such that no single failure will result in loss of power to both buses. In the event of loss of power to one bus, CSAS is initiated when required by the two measurement channels associated with the unaffected bus. Thus, no single failure can prevent proper CSAS actuation when required.

Each CSAS actuation channel can also be initiated manually from the control room.

The diverse containment heat removal function is also performed by the containment cooling system which is actuated by SIAS.

The containment isolation signal (CIS) is actuated on high containment pressure (5 psig) or high containment radiation (10 R/hr). The CIS measurement channels include four independent containment pressure transmitters and four independent containment radiation monitors. The measurement channels for each of these two diverse parameters are combined into a two-out-of-four logic. The output signals from the high containment and high radiation logic matrices are combined in an "or" logic circuit to form the containment isolation signal (CIS). There are two independent CIS actuation channels (A and B). The CIS channels close all the containment isolation valves that are not required for operation of the engineered safety features following an accident. In addition, per NRC Post TMI requirement, CIS is actuated by a Safety Injection Signal (SIAS).

Each CIS actuation channel may be initiated manually from the control room.

The function of the Main Steam Isolation Signal (MSIS) is to terminate blowdown of steam from the steam generators and normal feedwater flow to the steam generators in the event of a steam line break accident, and to effect isolation of the main steam lines in event of a LOCA.

The MSIS measurement channels include four steam generator pressure transmitters for each steam generator. The signals from the four sensors for each steam generator are combined in a two-out-of-four logic to provide closure of both main steam isolation valves, both main feedwater isolation valves, and both main feedwater pump discharge valves on low steam generator pressure (485 psig). The four feedwater valves also close on SIAS. The measurement channels (logic and actuation channels) associated with steam generator A are separated from those associated with steam generator B.

9.0 COMPONENT COOLING WATER SYSTEM (CCWS)

The component cooling system is designed to:

a) provide a heat sink for auxiliary systems under normal operating and shutdown conditions

b) provide an intermediate barrier between the reactor coolant and the intake cooling water systems

c) provide a heat sink for safety related components associated with reactor decay heat removal for safe shutdown or LOCA conditions, assuming a single failure

d) withstand design basis earthquake loads, tornado loads or maximum flood levels without loss of safety function

e) permit periodic inspection and testing of components to assure system integrity and capability

The component cooling system is a closed loop cooling water system that utilizes demineralized water (with potassium chromates, a corrosion inhibitor) to cool various components as shown schematically in Figure A.3. The component cooling system consists of two heat exchangers, three pumps, one surge tank, a chemical addition tank, and associated piping, valves, and instrumentation.

The component cooling system is arranged as two redundant essential supply header systems (designated A and B) each with a pump and heat exchanger and the capability to supply the minimum safety feature requirements during plant shutdown or LOCA conditions. The nonessential supply header (designated N) which is connected to both essential headers during normal operation is automatically isolated from both by valve closure on a safety injection actuation signal (SIAS). During normal operation, the nonessential header supplies cooling water to the following components: fuel pool heat exchanger, sample heat exchangers, boric acid concentrators, waste concentrator,

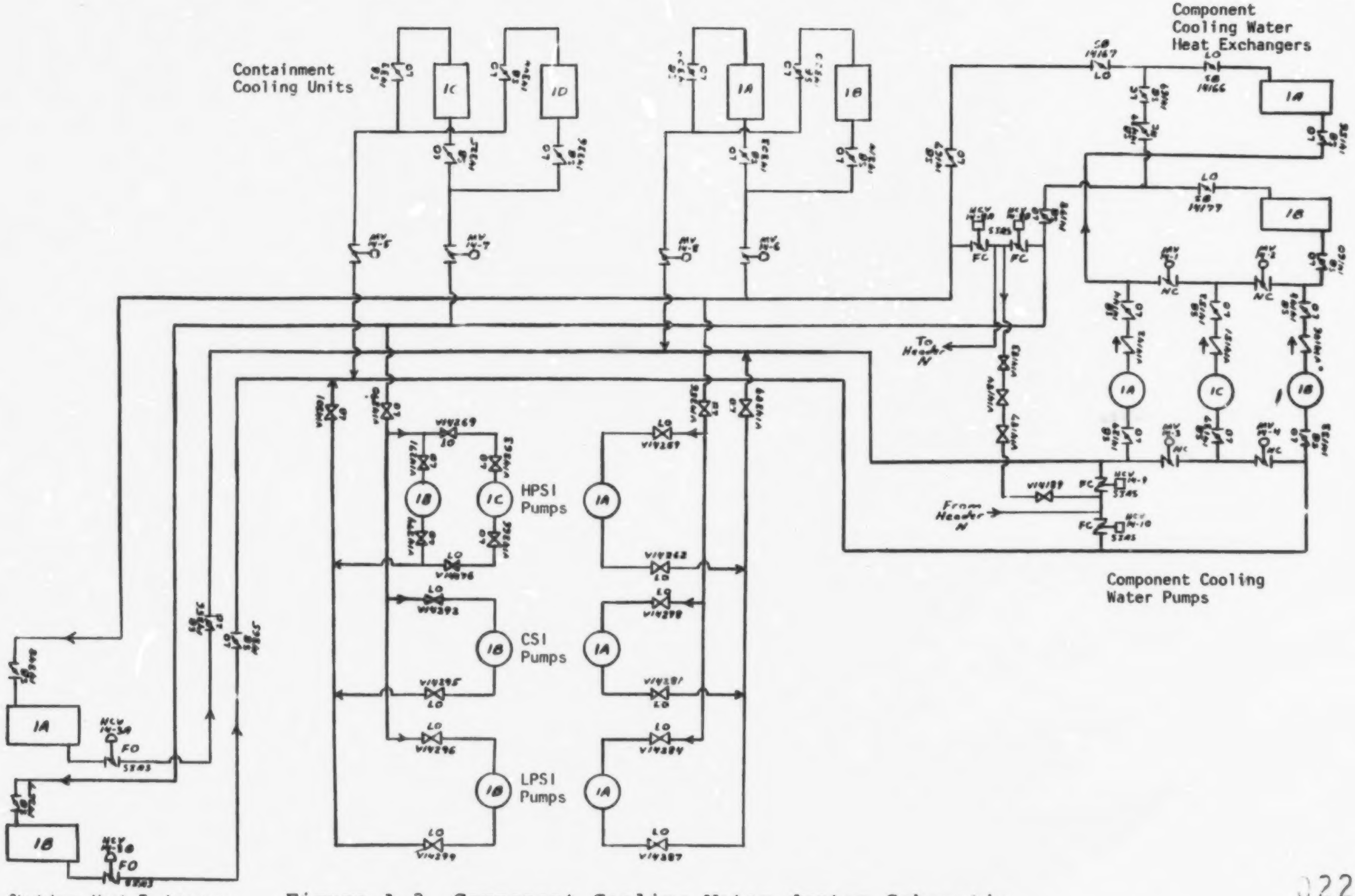
Containment Cooling Units
1C
1D
1A
1B
Component Cooling Water Heat Exchangers
1A
1B
SB 14167
SB 14166
SB 14177
LO
LC
MV 14-5
MV 14-7
MV 14-8
MV 14-6
MV 14-1
MV 14-2
MV 14-3
MV 14-4
HCV
FC
NC
To Header N
From Header N
V14189
HCV 14-9
HCV 14-10
1A
1C
1B
Component Cooling Water Pumps
V14269
V14276
1B
1C
HPSI Pumps
1A
V14267
V14262
V14292
V14295
1B
CSI Pumps
1A
V14298
V14281
V14296
V14299
1B
LPSI Pumps
1A
V14284
V14287
1A
1B
HCV 14-3A
HCV 14-8
FO
Shutdown Heat Exchangers

Figure A.3 Component Cooling Water System Schematic

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Figure A.3 waste gas compressors, letdown heat exchanger, control element drive mechanism air coolers, reactor coolant pump motors, and quench tank cooling units.

The A and B headers serve the following components:

Header A	Header B
Shutdown heat exchanger 1A	Shutdown heat exchanger 1B
Containment fan cooler 1A	Containment fan cooler 1C
Containment fan cooler 1B	Containment fan cooler 1D
Low pressure safety injection pump 1A	Low pressure safety injection pump 1B
High pressure safety injection pump 1A	High pressure safety injection pump 1B
Contain spray pump 1A	Containment spray pump 1B

High pressure safety injection pump 1C is suppied by header B by administrative procedures.

The A and B header systems are isolated from each other during accident conditions. Pump 1A serves header A and Pump 1B serves header B. Pump 1C may be aligned with either header A or B by means of the cross connection valving on the suction and discharge sides of the pumps and any misalignment between the component cooling water pump C motor power and the pump's motor valves will be annunciated in the Control Room. Figure A.3 shows the valve arrangement for pumps.

Both the A and B supply header systems pump demineralized cooling water through the shell side of their respective component cooling heat exchangers, through the components being cooled and back to their respective pumps. The surge tank is connected to the suction side of the pumps and is designed to accommodate volumetric thermal expansion and contraction in the system and to maintain a static pressure head at each pump suction. Demineralized makeup water is added to the surge tank through an automatic level control system by the demineralized water pump. Provisions are also made to supply makeup from the fire protection system.

The component cooling system is capable of providing sufficient cooling capacity to cool reactor coolant system and auxiliary systems components with two pumps and one heat exchanger in operation, although, during normal operation, flow is established through both heat exchangers. Two pumps and two heat exchangers are used during normal plant shutdown; however, if only one heat exchanger is available, the cooldown rate is decreased but plant safety is not jeopardized.

Safety related equipment cooling requirements are met following a postulated LOCA with only one pump and one heat exchanger operating, even though both essential header systems are available.

The component cooling pumps are connected to separate emergency electrical buses which can be energized by the diesel generators. The A and B pumps are connected to the respective A or B emergency bus counterparts and the third pump is connected to the AB bus which is remote manually connected from the control room to either the A or B diesel generator bus.

The component cooling system pumps, heat exchangers and portions of the piping and valves are located outdoors. These components are designed to operate under the following environmental conditions: ambient air temperature from 30°F to 120°F, 100% humidity, salt laden atmosphere, torrential rains and hurricane winds.

Other system components are located within the reactor auxiliary building and are designed for the normal ambient conditions expected in the building. No system active components are located in areas of high radioactivity.

Except for seismic Class I piping serving the containment fan coolers, there are no essential components located within the containment vessel. Containment post-accident environment will not affect operation of the component cooling system.

All components of the component cooling system including cooling water pumps, heat exchangers and piping which are essential for safe shutdown or to mitigate the effects of a LOCA are designed and installed as seismic Class I equipment. The nonessential portions of the system are not seismic Class I and are isolated automatically from the seismic Class I portions upon SIAS. The isolation valves are seismic Class I.

The pumps and heat exchangers are located outdoors on a seismic Class I foundation. Seismic qualification of system components has been demonstrated by manufacturer calculations. The pumps are separated by 18 feet from centerline to centerline. In addition, tornado missile shielding is provided so that the design basis missile will not preclude the availability of a pump. Indoor components such as the surge tank and portions of the piping and valves are protected from tornado winds and missiles since they are housed within the reactor auxiliary building.

Component cooling system equipment susceptible to flood damage is protected by locating all safety-related components above

0224

the maximum expected PMH water level and wave runup. The elevation of the maximum expected wave runup is 17.2 ft above mean low water.

10.0 INTAKE COOLING WATER SYSTEM (ICWS)

The intake cooling water system is designed to:

a) provide a heat sink for the component cooling, turbine cooling, and steam generator open blowdown cooling systems under normal operating and shutdown conditions

b) provide a heat sink for the component cooling system under safe shutdown or LOCA conditions, assuming a single active component failure

c) withstand design basis earthquake loads, tornado loads or maximum flood levels without loss of safety function

d) permit periodic inspection and testing of equipment to assure system integrity and capability.

The intake cooling water system is shown in Figure A.4. The system consists of three pumps and associated piping and valves. The system removes heat from the component cooling heat exchangers, the turbine cooling heat exchangers, and the steam generator open blowdown heat exchangers and discharges it to the condenser discharge canal. Intake cooling water from the intake structure flows through basket strainers located at the inlets of the component cooling, turbine cooling and steam generator open blowdown heat exchangers, passes through the tube side of the exchangers, and flows to the discharge canal.

The intake cooling water system is divided into two redundant supply header systems designated A and B. Both header systems, each aligned with an intake cooling water pump, supply normal plant operating and shutdown requirements. However, during accident conditions, one pump and header is adequate to supply the required cooling water to one component cooling heat exchanger. The turbine cooling water and open blowdown heat exchangers are supplied by nonessential headers which are automatically isolated on SIAS by valves I-MV 21-2 and 21-3. These valve operators are part of the ESFAS equipment and under local reopening after a postulated accident would produce an alarm and valve open indication in the control room. Under administrative control and procedures the control room operator would reclose the valves from the control room in such a case.

In the event that either pump 1A or 1B fails, intake cooling pump 1C may be aligned with either header A or B by positioning of the pump discharge header cross connect valve. Pump 1C has only manual (non-motor-operated) valves in the discharge

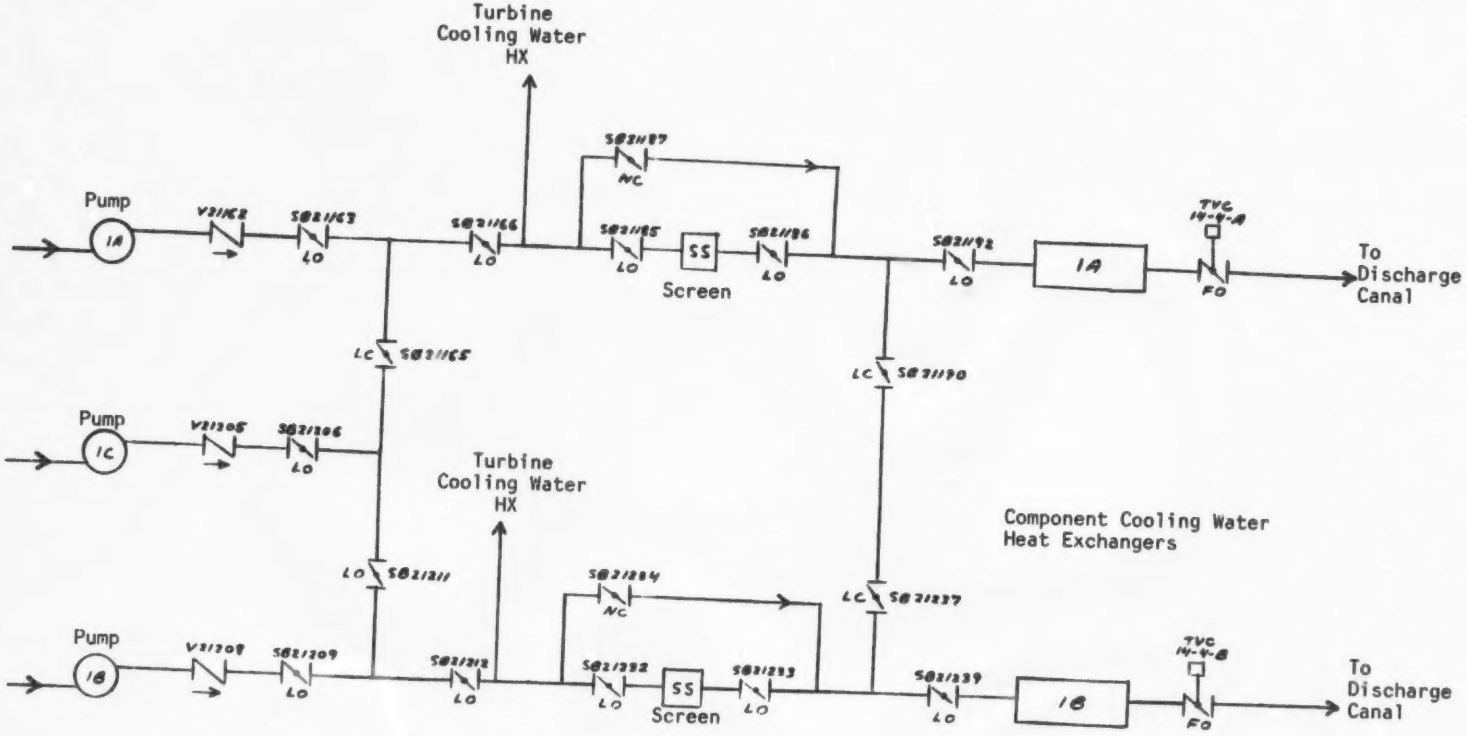
Turbine
Cooling Water
HX
Pump
1A
V21/62
SB21/63
LO
SB21/66
LO
SB21/87
NC
SB21/85
LO
SS
Screen
SB21/86
LO
SB21/92
LO
1A
TVC
14-4-A
FO
To
Discharge
Canal
LC
SB21/65
LC
SB21/90
Pump
1C
V21/205
SB21/206
LO
Turbine
Cooling Water
HX
Component Cooling Water
Heat Exchangers
LO
SB21/211
SB21/234
NC
LC
SB21/237
Pump
1B
V21/208
SB21/209
LO
SB21/212
LO
SB21/232
LO
SS
Screen
SB21/233
LO
SB21/239
LO
1B
TVC
14-4-B
FO
To
Discharge
Canal

Figure A.4 Intake Cooling Water System Schematic

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0226

header. A proper valve alignment is assured by a Technical Specification requirement.

The intake cooling water pumps supply their own bearing lube water from the discharge header. The domestic water system supplies lubrication only during initial pump startup.

The intake cooling water system is designed to supply sufficient cooling water to the component cooling heat exchangers to fulfill emergency requirements in the event of the design basis LOCA. Each essential header, with one intake cooling water pump operating, fulfills the requirements of one component cooling heat exchanger during normal, shutdown, and LOCA conditions.

The intake cooling water system is sized to ensure adequate heat removal with a design seawater temperature of 95°F at plant rated power.

The only time an interruption of bearing lube water supply can occur is after a loss of offsite power. In such a situation, the intake cooling water pumps are automatically loaded from the diesel generators. The interruption in operation of the intake cooling water pumps will be less than 30 seconds. The pumps can be restarted without damage to the bearings for at least the first minute following such an incident. If any of the pumps are running at this time, sufficient water will be present to provide lube water for all the pumps. The domestic water system is not needed for normal operation or restart of the intake cooling water pumps.

The redundant essential headers are isolated from each other during emergency conditions by two valves (I-SB-21-7) in the tie line connecting the headers. Each essential header is isolated from the nonessential portion of the system by a valve which closes automatically on SIAS. Each header isolation valve receives a signal from a separate SIAS channel, hence, no single failure can cause both valves to remain open.

The intake cooling water pumps and valves are located outdoors and are designed to operate under the following environmental conditions: ambient temperature from 30°F to 120°F, 100% humidity, salt laden atmosphere, torrential rains and hurricane winds.

The intake cooling water system pumps and valves, which had been specified in their respective purchase specifications as being suitable for outdoor service, had been installed in place for approximately two years prior to operation. Thus, the preoperational testing for these components demonstrated the adequacy of pump and valve operations after about two years exposure to the environmental conditions specified above.

Although it was felt that this exposure prior to commercial operation provided adequate assurance of operability, the Unit 1 motors and valves were operated while being sprayed with a fire hose as part of the Unit 1 preoperational test program.

The essential components of the intake cooling water system, i.e., those supplying cooling water to the comkponent cooling heat exchangers, are designed and installed as seismic Class I equipment. That portion of the system supplying cooling water to the turbine cooling water and steam generator open blowdown systems are non-seismic Class I downstream of the isolation valves. The seismic Class I and non-seismic Class I portions of the system are automatically isolated from each other on SIAS. The intake cooling pumps are located on the intake structure which is designed as seismic Class I. All underground piping is located in seismic Class I fill areas. Seismic qualification of system components are demonstrated by manufacturer calculations based on the design basis earthquake accelerations.

The equipment of the intake cooling water system is located outdoors. Accordingly, the essential portions of the system are designed to withstand design basis tornado winds of 360 mph and PMH winds of 194 mph. Atmospheric depression resulting from these winds is 3.0 psi and 1.5 psi, respectively. The intake structure housing these pumps is designed as seismic Class I and can withstand the effects of winds and wave action resulting from the PMH tornadoes.

The manually operated intake cooling water header valves are not affected by hurricane-induced sprays. Isolation valves I-MV-21-2 and 3 are powered by Limitorque motor operators with weatherproof enclosure and thus suitable for outdoor service. The valves are located in the valve pit adjacent to the pump structure to protect them from water spray. The valve and motor pit structure is designed for tornado wind loading.

Protection is provided from missiles by imposing the following spatial separation criteria on redundant components located outdoors:

a) Minimum separation for piping up to a height of 25 ft above yard grade is controlled by the dimensions of an airborne car weighing 4000 lb, 18 ft long, 7 ft wide, travelling at 50 mph; for piping at elevation higher than 25 ft above yard grade, minimum separation is controlled by a 2 in. by 4 in. timber, 10 ft long, travelling at 360 mph.

b) Underground piping minimum separation is 8.7 ft, dictated by tornado missiles. A minimum of 6 ft earth or equivalent concrete cover is provided for protection from the airborn car.

0228

The intake cooling water pumps are physically separated and provided with tornado missile shielding, so that the design basis missile will not preclude the availability of one pump which is the minimum requirement for safe shutdown. The physical separation of the pump power supply is consistent with the pump physical separation.

To ensure pump operation under flood conditions, the pump motors are flood protected to elevation 22.8 ft which is above the maximum calculated flood level. The pump suction columns require 4 ft of minimum submergence to deliver the design capacity of the pumps. This requirement is met under the minimum water level conditions associated with the maximum probably hurricane postulated for the site.

11.0 ELECTRIC POWER SYSTEM (EPS)

The electric power system provides both AC and DC power to the various components and instruments within the plant. The emergency or safeguards EPS, which is modeled in this analysis, is that portion of the EPS which provides both AC and DC electric power to those vital systems required to operate during an accident. The EPS, as modeled, is shown on Figure A.5.

The normal source of auxiliary AC power for plant start-up or shutdown is from the incoming offsite transmission lines through the plant switchyard and start-up transformers. The start-up transformers step down the 240 kV incoming line voltage to 6.9 kV and 4.16 kV for auxiliary system use.

During normal plant operation, AC power is provided from the main generator through the unit auxiliary transformers. The unit auxiliary transformers step down the main generator output voltage from 22 kV to 6.9 kV and 4.16 kV.

Normal transfer of the 6.9 kV or 4.16 kV auxiliary buses between the unit auxiliary and start-up transformers will be initiated by the operator from the control room; emergency transfer from the auxiliary transformer to the start-up transformer will be initiated automatically by protective relay action.

In the event of a complete loss of the normal power sources, station onsite AC power will be supplied by the onsite emergency diesel generators and station batteries.

Each of the unit auxiliary and start-up transformers and each emergency diesel generator has sufficient capacity to supply the safety related loads for safe plant shutdown or to mitigate the consequences of a design basis accident.

Two 6.9 kV buses are provided, each supplied from a unit auxiliary or alternatively from a start-up transformer. Each

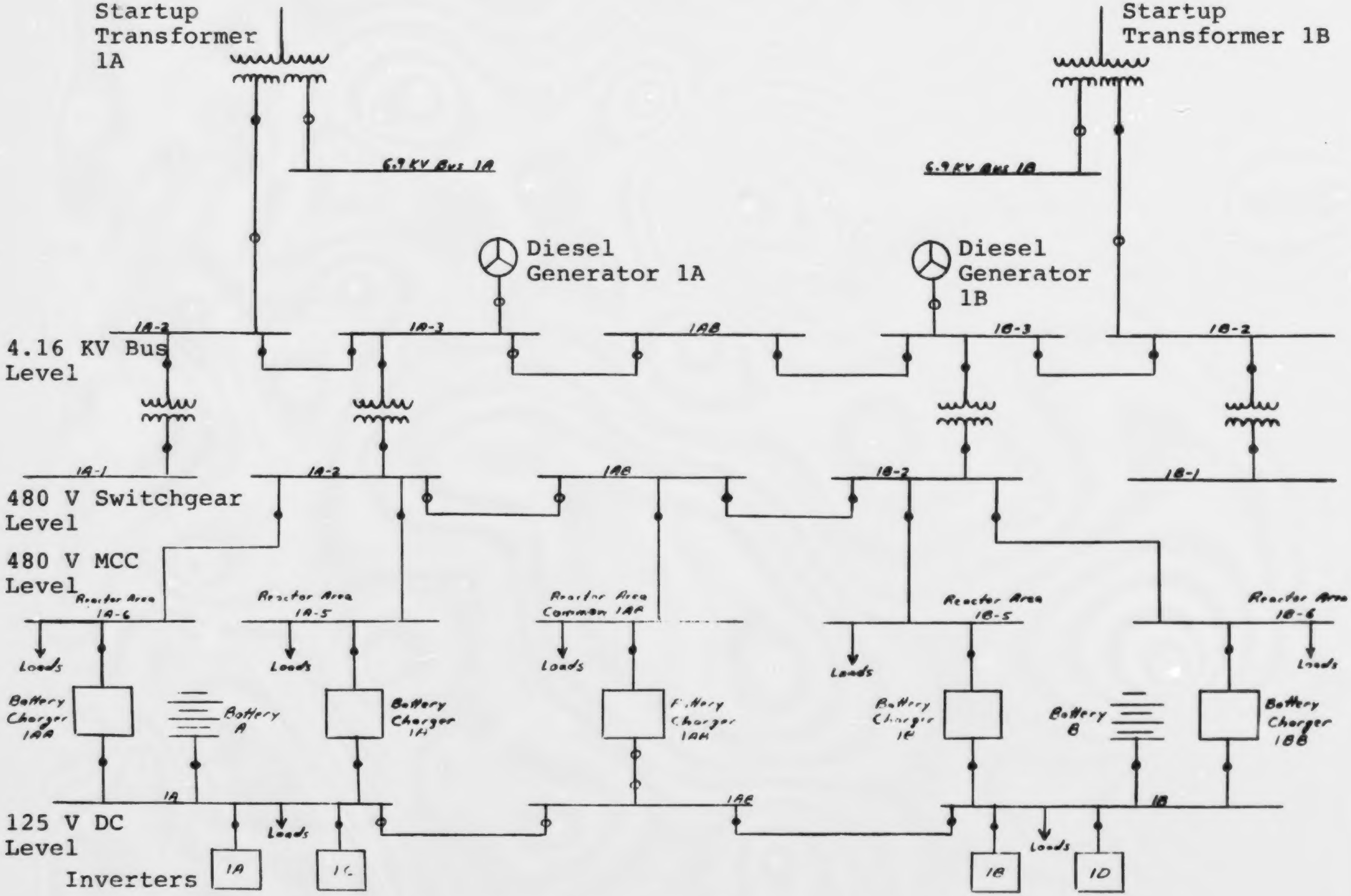
Startup Transformer 1A
Startup Transformer 1B
6.9 KV Bus 1A
6.9 KV Bus 1B
Diesel Generator 1A
Diesel Generator 1B
4.16 KV Bus Level
1A-2
1A-3
1AB
1B-3
1B-2
480 V Switchgear Level
1A-1
1A-2
1AB
1B-2
1B-1
480 V MCC Level
Reactor Area 1A-6
Reactor Area 1A-5
Reactor Area Common 1AB
Reactor Area 1B-5
Reactor Area 1B-6
Loads
Loads
Loads
Loads
Loads
Battery Charger 1AA
Battery A
Battery Charger 1A
Battery Charger 1AB
Battery Charger 1B
Battery B
Battery Charger 1BB
1A
1AB
1B
Loads
Loads
125 V DC Level
Inverters
1A
1C
1B
1D

Figure A.5 Onsite Electric Power System Schematic

0230

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6.9 kV bus serves two reactor coolant pump motors and one steam generator feedwater pump motor. The 6.9 kV buses are non-safety related, or normal buses, since they serve equipment that is not required to function for safe plant shutdown or for design basis accidents. Hence, they do not require backup from the emergency power system.

The 4.16 kV system consists of normal and emergency buses. Normal buses provide power to loads which are non-safety related. All safety related loads are powered from the emergency buses. There are two normal buses which receive power directly from the unit auxiliary or start-up transformers.

The emergency portion of the 4.16 kV system is arranged into two redundant load groups (A and B). Each of these load groups consists of the complement of safety related equipment needed to achieve safe plant shutdown and/or to mitigate the consequences of a design basis accident. Additional safety related equipment (e.g., the third component cooling and intake cooling water pumps) is arranged as a "third service" or "swing" load group AB. This load group consists of equipment which can be used as backup or replacement to the equipment in either of the main redundant load groups A or B. Load group A is powered by the emergency bus 1A-3 and load group B is provided by emergency bus 1B-3. Emergency bus 1AB serves load group AB.

Whenever the normal power sources are available, power is supplied directly to the 4.16 kV system through the two normal buses, each of which is tied to one of the redundant emergency buses. Emergency bus 1AB is normally tied to either (but never both) emergency bus 1A-3 or 1B-3. The tie breakers of bus 1AB are interlocked electrically to prevent this bus from being simultaneously connected to both buses 1A-3 and 1B-3.

Upon a loss of the normal power sources, the tie breakers between the normal and emergency buses will automatically open and the emergency diesel generators will automatically begin supplying power directly to the emergency buses. The pressurizer heater transformers, which are a non-safety related load, are supplied from the emergency 4.16 kV emergency buses. The loads are tripped from these buses upon loss of offsite power, however, and can only be reconnected to the buses manually.

The 4.16 kV normal buses 1A-2 and 1B-2 are rated 3000 amps and are provided with 3000 amp incoming breakers of 350 MVA interrupting capacity. Emergency buses 1A-3, 1B-3, and 1AB are rated 1200 amps. The incoming feeder breakers of these last three buses and all outgoing breakers of all five 4.16 kV buses are 1200 amps, 250 MVA interrupting capacity and 80,000 amp momentary duty.

0231

The arrangement of the 480 volt system is similar to that of the 4.16 kV system with buses designated as normal or emergency. There are two normal 480 volt buses each powered by one of the normal 4.16 kV buses through a station service transformer. There are no interconnections between the normal and emergency portions of the 480 volt system.

The emergency portion of the 480 volt system is also arranged into redundant load groups A and B served by 480 volt buses 1A-2 and 1B-2 respectively with a swing load group AB served by 480 volt bus 1AB. Power is transmitted from the 4.16 kV emergency buses through the station service transformer to 480 volt buses 1A-2 and 1B-2. The swing bus 1AB is normally tied to either one of the redundant emergency 480 volt buses. All the AB buses (4.16 kV, 480 volt, and 125 volt DC) will be connected to either the corresponding A buses or B buses at any one time. That is, for example, the operation of the 4.16 kV bus AB connected to the 4.16 kV bus 1A-3 and the 480 volt bus AB connected to the 480 volt bus 1B-2, will not be permitted. Alarms are provided to alert the operated if the AB buses on all voltage levels are not aligned properly.

The 480 volt buses supply power directly to motors rated above 100 HP, but generally below 300 HP, and to 480 volt motor control centers (MCC) throughout the plant. The MCCs provide power to motors rated below 100 HP. Emergency MCCs are powered from emergency power centers while normal MCCs are powered from normal power centers. A "third service" emergency MCC is provided which is fed from 480 volt bus 1AB.

Emergency 480 volt bus 1AB may be connected to either (but never both) bus 1A-2 or bus 1B-2 through 1600 amp breakers with delayed trips. There are two breakers in series in each tie which are electrically interlocked to prevent the 1AB bus from being simultaneously connected to bus 1A-2 and 1B-2. The short circuit level of this section is 30,000 amp symmetrical. This section feeds a third charging pump and the emergency "third service" MCC 1AB through 6700 amp breakers.

There is a total of eighteen 480 volt motor control centers throughout the plant, eleven for normal service and seven for emergency service. All MCCs except 1AB are fed from the reduced short circuit level sections of the 480 volt switchgears as explained above. In no case are two redundant pieces of equipment connected to the same motor control center. For both normal and emergency services, there are two redundant MCCs in each area, each one connected to a redundant 480 volt switchgear. For normal vital services there is a "common" MCC 1C fed from either one of the two normal 480 volt switchgears through breakers at the switchgears and in an automatic transfer switch at the MCC. For emergency third service loads there is an MCC 1AB fed from the 480 volt switchgear 1AB. This last MCC is provided with a current

0232

limiting reactor to reduce its short circuit level down to 14,000 amp symmetrical. All other MCCs in the plant also have a short circuit level of 14,000 amp symmetrical but do not require reactors since they are fed from the short circuit low level (after the reactors) section of the 480 volt switchgear buses.

In some cases, there are non-emergency loads connected to emergency MCCs. Whenever this occurs, the MCC bus is split into a vital and a non-vital section connected through a bus isolating contactor that automatically drops open during an emergency, thus separating the non-emergency loads from these MCCs. These non-emergency loads consist of normal lighting, normal power panels and power receptacles.

The standby AC power source consists of two redundant diesel generator sets, their attendant air starting and fuel supply system, and automatic control circuitry.

The diesel generator sets supply power to those electrical loads which are needed to achieve safe shutdown of the plant or to mitigate the consequences of a loss of coolant accident (LOCA) in the event of a coincident loss of normal AC power supply. The power required for the accident condition is higher by a large margin over that required for shutdown.

Each diesel generator set consists of two diesel engines mounted in tandem with a 3500 kW generator coupled directly between the engines.

Each engine in each diesel generator set has a self-contained cooling system which consists of a forced circulation cooling water system which cools the engine directly and an air-cooled radiator system which removes the heat from the cooling water. The cooling water pump and radiator fan are driven directly from the engine crankshaft. The diesel generator set system requires no external source of power and does not depend on any plant cooling system.

The engine of each diesel generator set has a self-contained lube oil system consisting of a lube oil sump located at the base of the engine, a lube oil pump, piping, and a heat exchanger. The lube oil heat exchanger is served by the diesel generator set cooling water system. No external source of power or other plant system is required for the diesel generator set lube oil system.

In the event of loss of normal sources of power to the onsite power system, each diesel generator set is automatically started and loaded by controls and circuitry which are independent of that used to start and load the redundant unit.

The DC power system is also shown on Figure A.5. Power is provided at 125 volts DC (ungrounded) for plant control and instrumentation and for operation of DC motor operated equipment such as valve operators and emergency lube oil pumps. As for the 4.16 kV and 480 volt AC emergency systems, the 125 volt DC system is arranged into two main redundant load groups A and B and a third service or swing load group AB. Load groups A and B are each capable of supplying the minimum DC power requirements to safely shut down the plant and/or mitigate the consequences of a LOCA.

Load group A is served by DC bus 1A and load group B by DC bus 1B. Load group AB is served by DC bus 1AB which is normally tied to either (but never both) DC bus 1A or 1B corresponding to the manner in which the 4.16 kV and 480 volt AB buses are connected to their respective A or B buses. These are two breakers in series in each tie which are key interlocked to prevent the 1AB bus from being simultaneously connected to both the 1A and 1B buses.

Each 125 volt DC bus will be supplied from one of the 125 volt DC battery chargers (300 amp capacity each), two of which are normally operating. Each charger is sized to carry normal DC load and to recharge a battery from 1.75 volts per cell. The worst loading condition on the battery chargers occurs during post LOCA condition with loss of offsite power. After the initial 40 second battery operation, the battery charger is sufficient to recharge its battery within a 1/2 hour period. Two 125 volt DC battery chargers operate separately on two of the buses. A third 125 volt DC battery charger on the AB bus provides a backup for the two operating 125 volt DC chargers.

Each of the two 25 volt lead-calcium type station safety batteries consists of 60 cells and is rated 1800 amp-hr at 8 hour discharge rate. The above rating is sufficient for an 8-hour emergency period without assistance from a battery charger with all non-emergency loads disconnected within one hour.

The battery chargers and inverters are automatically loaded on the diesel generator approximately 40 seconds after loss of offsite power, thus returning the DC system to normal. The above 8 hour battery rating is more than adequate for this design limiting case of 40 second battery operation.

Since the station batteries are normally connected to the essential distribution panels for charging, they will maintain voltage to the 125 volt DC system in the event of loss of the battery chargers as a power source. Thus, upon a loss of normal power, there will be no discontinuity of DC power for protective system and emergency diesel generator control circuitry. The station batteries have more than enough capacity to supply all essential DC power until the battery chargers can be restored on emergency or normal AC power source.

Appendix B

INTERNAL ANALYSIS FOR THE ST. LUCIE NUCLEAR POWER PLANT

by

Gary A. Sanders
Sandia National Laboratories

0235

TABLE OF CONTENTS

1.0 INTRODUCTION B-3 237

2.0 EVENT TREES AND SYSTEM SUCCESS CRITERIA B-4 238
2.1 Transient Event Trees B-4
2.1.1 Transient System Success Criteria B-5 239
2.1.2 T_1 - Loss of Offsite Power (LOP) Transient Event Tree B-6 240
2.1.3 T_2 - Loss of Feedwater Transient Event Tree B-8 242
2.1.4 T_4 Loss of AC Bus, T_5 Loss of DC Bus B-8 242
2.1.5 T_3 - All Other Transients B-8 242
2.2 Small LOCA Event Tree B-10 244
2.2.1 Small LOCA System Success Criteria . . . B-10 244
2.2.2 Small LOCA Event Tree B-13 247
2.3 Containment Systems Event Tree and Success Criteria B-14 248
2.3.1 Containment Systems Success Criteria . . B-16 250
2.3.2 Containment Systems Event Tree B-16 250

3.0 DATA BASE B-17 521
3.1 Initiating Events and Events Not Modeled B-17 251
3.2 Basic Event Probabilities B-23 257
3.3 Failure Probability Rationale B-30 264
3.3.1 Battery Failure Rate B-30 264
3.3.2 Local Control Circuit Failures B-30 264
3.3.3 Unavailability Due to T&M Downtime . . . B-31 265
3.3.4 Common Mode Failure B-32 266

4.0 RECOVERY ANALYSIS B-35 269

5.0 ACCIDENT SEQUENCE ANALYSIS B-41 275
5.1 Accident Sequence Delineation B-42 276
5.2 Accident Sequence Quantification B-45 279
5.2.1 Quantification including Bleed and Feed and Secondary Blowdown Capability B-54 288
5.2.2 Quantification Given Secondary Blowdown Not Applicable B-54 288
5.2.3 Quantification Given Bleed and Feed Not Applicable B-68 302
5.2.4 Long Term Station Blackout Analysis . . . B-68 302
5.2.5 Quantification Given PORV Block Valves are Normally Open B-74 308
5.3 Vulnerability Identification B-76 310
5.4 Containment System Integration B-91 325

6.0 VALUE ANALYSIS B-94 328
6.1 Core Melt Probabilities B-94 328
6.1.1 Modification Values B-94 328
6.1.2 Alternative Values B-97 331
6.2 Containment Systems Probabilities B-100 334

REFERENCES B-102 336

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1.0 INTRODUCTION

The internal analysis involves a number of sequential steps which are described in this appendix. Each step and the section where it is discussed in the appendicies or the main report is shown below.

1) Establish System Success Criteria (Appendix B, Section 2.0)

2) Construct Event Trees (Appendix B, Section 2.0)

3) Construct Fault Trees (Appendix A)

4) Establish Data Base (Appendix B, Sections 3.1 and 3.2)

5) Visit Plant for System and Special Emergency Information (Appendix B, Section 5.0)

6) Delineate Accident Sequences (Appendix B, Section 5.1)

7) Perform Core Melt Analysis Using the SETS[1] Computer Program (Appendix B, Section 5.2)

8) Determine Core Melt Dominant Accident Sequences (Appendix B, Section 5.2)

9) Perform Recovery Analysis on Core Melt Dominant Accident Sequences (Appendix B, Section 5.2 and Section 4.0 of the Main Report)

10) Identify Individual Vulnerabilities from Dominant Accident Sequence Cut Sets (Section 4.0 of Main Report and Appendix B, Section 5.2)

11) Propose Modifications for the Most Significant Vulnerabilities (Section 4.0 of Main Report)

12) Integrate Internal and Special Emergency Vulnerabilities (Section 4.0 of Main Report)

13) Determine Potential Alternatives (Section 4.0 of Main Report)

14) Communicate Alternatives to Architectural Engineer (Section 4.0 of Main Report)

15) Visit Plant to Verify Modification Design (Section 4.0 of Main Report)

16) Perform SETS Computer Program Analysis for Combined Core Melt Dominant Accident Sequence and Containment System Sequences (Appendix B, Section 5.4)

17) Reapply Recovery Analysis to Include the Containment Systems (Appendix B, Section 5.4 and Section 4.0 of the Main Report)

18) Assign Dominant Accident Sequences Including Containment System and Containment Failure Mode Probabilities (Appendix B, Section 5.4)

19) Reevaluate Core Melt Probabilities for Each Alternative (Appendix B, Section 6.1)

20) Reevaluate Combined Core Melt and Containment Failure Probabilities and Release Category Assignments for Each Alternative (Appendix B, Section 6.2)

Sections 5.0 and 6.0 of Appendix B are the basic description of the internal analysis while Sections 2.0, 3.0, and 4.0 provide the necessary supporting information.

2.0 EVENT TREE AND SYSTEM SUCCESS CRITERIA

This Section of Appendix B describes the St. Lucie event trees and the associated system success criteria. These event trees are related in the following ways. Those branches of the transient event trees that represent transient induced LOCAs, i.e., stuck open pressurizer relief valves, connect to the appropriate branch of the LOCA event tree according to the event successes and failures preceding the induced LOCA. Also, since core melt is not affected by containment systems, the containment event tree can be separate and simply attached to the end of each core melt accident sequence for either the transient or LOCA event trees. These two relationships greatly simplify the event tree presentation and will be discussed further in Section 5.1 as Accident Sequence Delineation and in Section 5.3 on Containment Systems Integration.

2.1 Transient Event Trees

There were some transients which were not in the defined scope of work for this program, e.g., anticipated transients without scram (ATWS) and pressurized thermal shock (PTS). Those transients which are considered can be represented by three fundamental event trees as defined below:

T_1 - Loss of Offsite Power (LOP) Transient

T_2 - Loss of Feedwater (LOF) Transient (offsite power is assumed to be available)

T_3 - All Other Transients (AOT) where the power conversion system is initially available (e.g., Turbine Trip)

0238

Two support system initiating events that will use the same structure as the T_2 event tree are:

T_4 - Loss of an AC bus which is assumed to trip the unit
T_5 - Loss of a DC bus which is assumed to trip the unit

There are several features that are common to all the transient event trees.

1) The "bleed and feed" (B&F) mode of primary system cooling is a viable option at St. Lucie, and is included in their revised emergency operating procedures. This mode of operation involves opening the PORVs to release steam and subsequent high pressure injection of RWST water for cooling and makeup. The "bleed" comes before "feed" since the high pressure (safety injection) pump shutoff head is substantially below the safety relief valve settings.

2) It is assumed that no containment overpressure protection would be required at St. Lucie for the sensitivity study of the B&F mode. This was confirmed by plant personnel and is supported by the IREP ANO-1 study[2] which showed that it takes over 70 hours of B&F without containment cooling before the containment integrity is challenged.

3) The Chemical Volume and Control System (CVCS), whose function is to cool the Reactor Coolant Pump (RCP) seals, is not included in the transient event trees. It is assumed that the RCP seals will not require cooling. This issue appears to be unresolved at the present time. Also, it is assumed that the CVCS will not be required for coolant makeup while cooling down to stable hot shutdown.

4) The reactor protection system (RPS) is assumed to succeed in each transient event tree.

2.1.1 Transient System Success Criteria

Each heading in the transient event tree relates to a system or function with an associated success criteria. In a few cases the system is not fault tree modeled, but it is always important to understand what equipment in the system is required for success. Although the support systems are not shown explicitly in the event trees, the success criteria are also given below.

System	Success Criteria
Power Conversion System (PCS) - M	1 of 2 trains (no fault tree model)
Auxiliary Feedwater Systems (AFWS) - L	1 of 2 Motor Driven Pump (MDP) Trains or the Turbine Driven Pump (TDP) Train

0239

Pressurizer Safety Relief Valves fail to open on demand (SRVO) - P	1 of 3 safety relief valves given the PORVs are blocked
Pressurizer Safety Relief Valve fails to reclose given one has opened (SRVR) - Q	It is assumed one safety relief valve opens and thus must reclose when the pressure decreases
Power Operated Relief Valves (PORV)	2 of 2 PORVs are required for B&F mode of operation. This is part of the B&F fault tree (E)
High Pressure Injection System (HPIS)	1 of 3 safety injection pump trains. This is part of the B&F fault tree (E). Operator initiation of the system is assumed
High Pressure Recirculation System (HPRS) - H_1	1 of 3 safety injection pump trains
Intake Cooling Water System (ICWS)	1 of 3 ICW pumps
Component Cooling Water System (CCWS)	1 of 3 CCW pumps and 1 of 2 heat exchangers

2.1.2 T_1 - Loss of Offsite Power LOP Transient Event Tree

The T_1 event tree is given in Figure B.1 with the accident sequences and the core status; OK indicating success or no core melt, late core melt (LCM) or early core melt (ECM), and LOCA for those sequences which transfer to the LOCA event tree. Only the core melt or transient induced LOCA accident sequences are numbered.

Success of the RPS is assumed. Failure of the PCS results from loss of offsite power. Thus, the significant part of the event tree starts with the success/failure of the AFWS. Given success of the AFWS, the SRVs are not required to open but may inadvertently open and then be required to reclose. Failure to reclose results in a transient induced LOCA where L is successful (Sequence 1). Successful reclosing is an OK sequence.

If the AFWS fails, the safety relief valves will be challenged and failure to open causes a rapid RCS overpressure and an assured reactor vessel rupture without further mitigation options resulting in core melt (Sequence 5). If, however, the safety relief valves open, the pressure is relieved and the

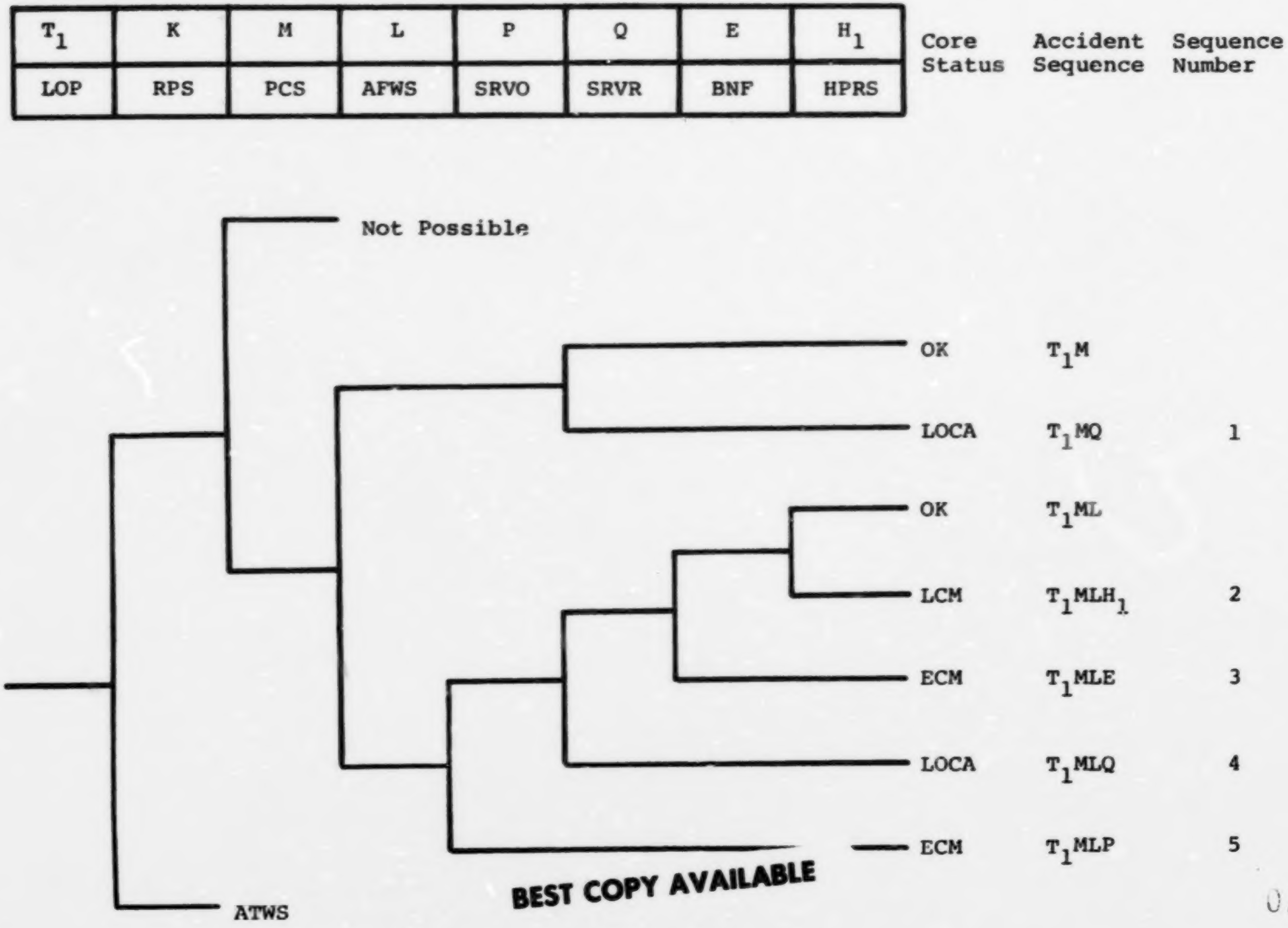
T_1
K
M
L
P
Q
E
H_1
LOP
RPS
PCS
AFWS
SRVO
SRVR
BNF
HPRS
Core Status
Accident Sequence
Sequence Number
Not Possible
OK
T_1M
LOCA
T_1MQ
1
OK
T_1ML
LCM
T_1MLH_1
2
ECM
T_1MLE
3
LOCA
T_1MLQ
4
ECM
T_1MLP
5
ATWS

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Figure B.1 Loss of Offsite Power Transient (T_1) Event Tree

next branch is success or failure of the safety relief valves closing. Failure of the SRVs to reclose results in a transient induced LOCA where L has failed (Sequence 4). For successful SRV reclosure, it is then necessary to consider another method to remove decay heat since the AFWS has failed previously in this scenario.

Bleed and Feed (B&F) is the next choice with failure leading to core melt (Sequence 3). Success of the B&F mode requires subsequent long term cooling and makeup from the HPRS. The high pressure components are the same, but both PORVs are also needed for pressure relief. Success of the HPRS results in an OK, no core melt sequence. Failure of the HPRS leads to core melt (Sequence 4).

2.1.3 T_2 - Loss of Feedwater Transient Event Tree

The T_2 event tree structure is exactly the same as the T_1 event tree. The only difference is the frequency of the initiating event and various system failures because of the initial conditions. The discussion of sequences in Section 2.2.2 applies to the T_2 event tree. The T_2 event tree is given in Figure B.2 for completeness.

2.1.4 T_4 - Loss of an AC Bus
T_5 - Loss of a DC Bus

T_4 and T_5 are two support system initiating events that will use the same structures as the T_2 event tree in Figure B.2.

T_4 is a loss of an AC bus which is assumed to trip the unit. In this case, the 1B-3 4160 volt AC bus was selected for analysis.

T_5 is a loss of a DC bus which is also assumed to trip the unit. The 1B DC bus was selected as the example case since failure of this bus would also fail the ability to start diesel generator 1B.

For both T_4 and T_5, initial loss of main feedwater is assumed but may be recovered. The only changes to the T_2 event tree are the initiating event frequency and the initial condition that a bus is unavailable.

2.1.5 T_3 - All Other Transients

This transient category is distinguished from T_1, T_2, T_4, and T_5 by the power conversion system being initially available. The T_3 transient addresses a large number of transients where the initial conditions are essentially the same, i.e., offsite power is available and the power conversion system is initially available. The event tree is given in Figure B.3.

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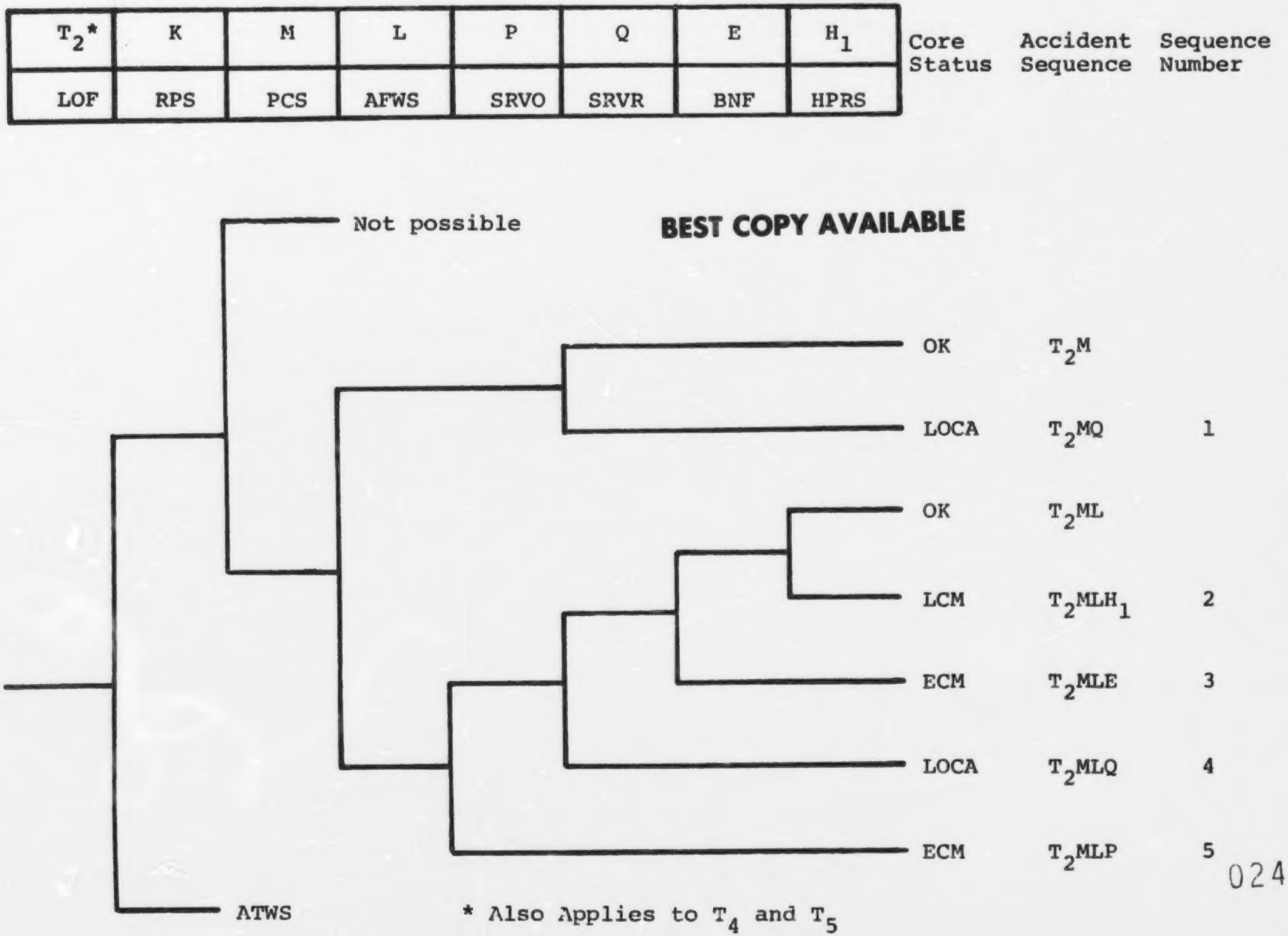
T2*
K
M
L
P
Q
E
H1
LOF
RPS
PCS
AFWS
SRVO
SRVR
BNF
HPRS
Core Status
Accident Sequence
Sequence Number
Not possible
OK
T2M
LOCA
T2MQ
1
OK
T2ML
LCM
T2MLH1
2
ECM
T2MLE
3
LOCA
T2MLQ
4
ECM
T2MLP
5
ATWS

* Also Applies to T_4 and T_5

Figure B.2 Loss of Feedwater Transient (T_2) Event Tree

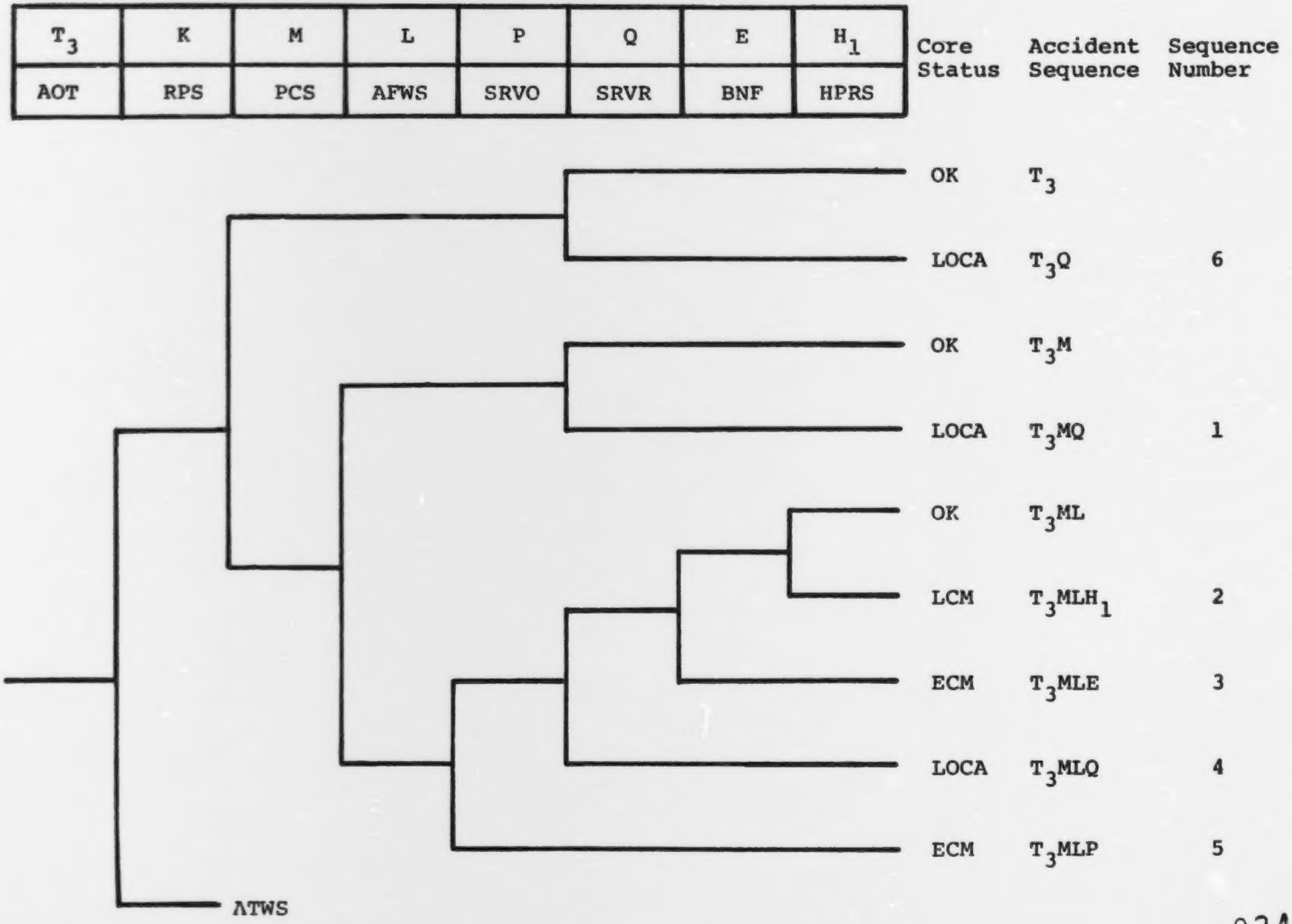
T3
AOT
K
RPS
M
PCS
L
AFWS
P
SRVO
Q
SRVR
E
BNF
H1
HPRS
Core Status
Accident Sequence
Sequence Number
OK
T3
LOCA
T3Q
6
OK
T3M
LOCA
T3MQ
1
OK
T3ML
LCM
T3MLH1
2
ECM
T3MLE
3
LOCA
T3MLQ
4
ECM
T3MLP
5
ATWS

Figure B.3 All Other Transients (T_3) Event Tree

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The discussion found in Section 2.1.2 for the T_1 transient is applicable except that the system failure frequencies will be different and there is one additional branch. This branch starts with PCS (M) success and thus the AFWS is not called upon. If one of the SRVs inadvertently opens, failure to reclose results in a transient induced LOCA (Sequence 6) which transfers to the LOCA event tree where a choice on L is required. Even though the PCS was initially available, it will trip on a safety injection signal for transient induced LOCAs but could be recovered if AFW failed. If the SRVs reclose, this becomes an OK sequence.

2.2 Small LOCA Event Tree

The small LOCA (< 2" diameter break) event tree for St. Lucie is shown in Figure B.4. It has two important features; a branch where credit is given for bleed and feed, and a branch where credit is given for secondary blowdown. These will be discussed further later in this section.

The defined scope of the program excludes large LOCAs (A), small LOCAs (S_1), reactor vessel rupture LOCAs (R), interfacing system LOCAs (V), and double steam generator tube ruptures. Single steam generator tube ruptures are considered as part of S_2 since that steam generator can be isolated. Small steam line breaks are also considered because they can either be isolated or will provide the secondary steam relief instead of the atmospheric relief valves or the turbine bypass valves.

2.2.1 Small LOCA System Success Criteria

Each heading in the LOCA event tree relates to a system or function with an associated success criteria. In a few cases the system does not have a fault tree model but success criteria are given below for all the systems implicitly or explicitly involved in the headings.

System	Success Criteria
Power Conversion System (PCS) - M	1 of 2 trains (no fault tree model)
Auxiliary Feedwater	1 of 2 motor driven pumps (MDP) or the turbine driven pump (TDP)
High Pressure Injection System (HPIS) - D_1	1 of 3 safety injection pump trains
Power Operated Relief Valves (PORV) - P_1	2 of 2 PORVs are required for B&F mode of operation

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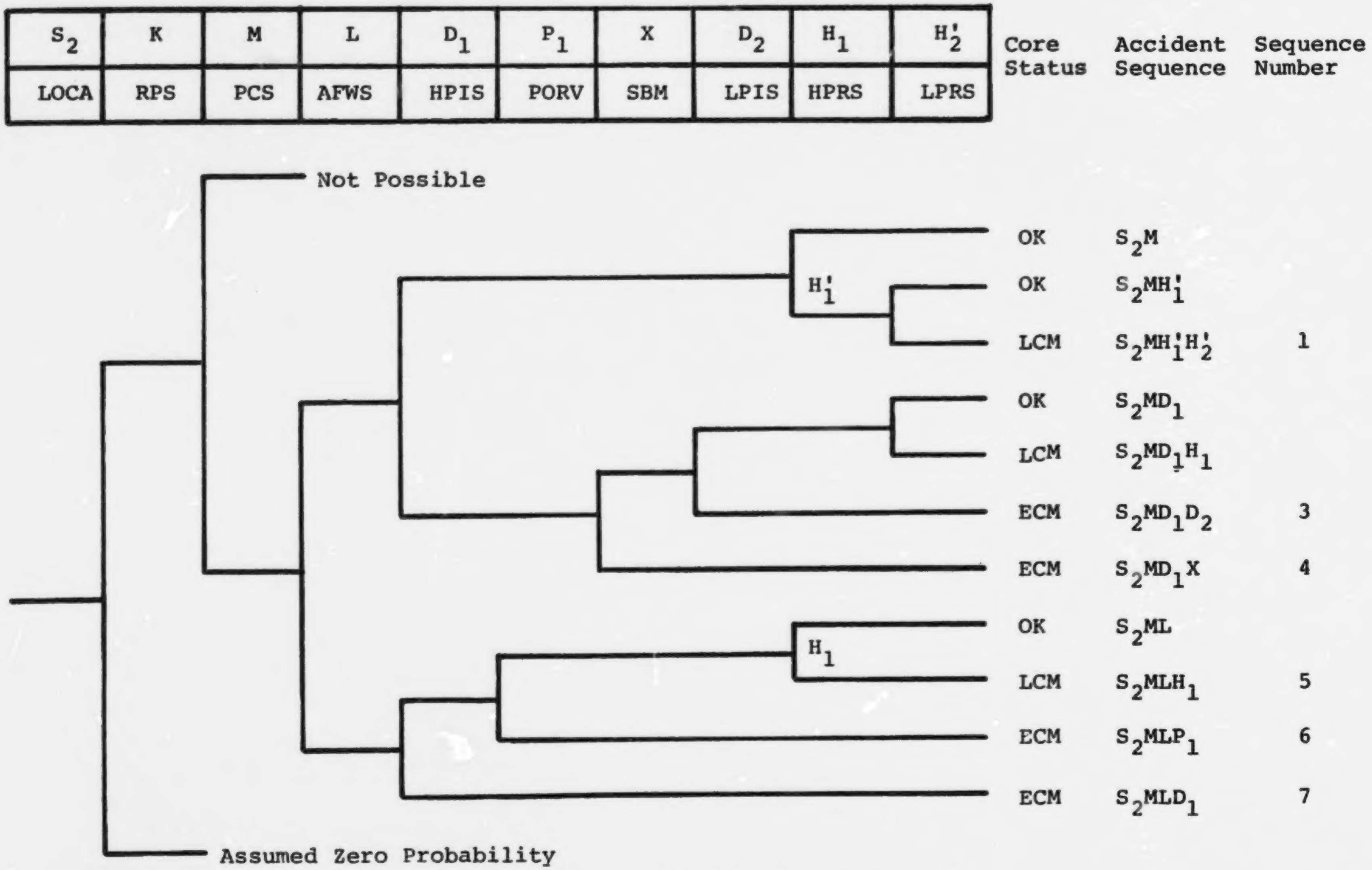
S_2
LOCA
K
RPS
M
PCS
L
AFWS
D_1
HPIS
P_1
PORV
X
SBM
D_2
LPIS
H_1
HPRS
H_2'
LPRS
Core Status
Accident Sequence
Sequence Number
Not Possible
H_1'
OK S_2M
OK S_2MH_1'
LCM $S_2MH_1'H_2'$ 1
OK S_2MD_1
LCM $S_2MD_1H_1$
ECM $S_2MD_1D_2$ 3
ECM S_2MD_1X 4
H_1
OK S_2ML
LCM S_2MLH_1 5
ECM S_2MLP_1 6
ECM S_2MLD_1 7
Assumed Zero Probability

Figure B.4 Small LOCA (S_2) Event Tree

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Secondary Blowdown Mode (SBM) - X	Normal pressurizer spray or auxiliary pressurizer sprays, and steam bypass valves or 2 of 2 atmospheric dump valves
Low Pressure Injection System (LPIS) - D_2	1 of 2 RHR pump trains
High Pressure Recirculation System (HPRS) - H_1	1 of 2 safety injection pump trains and 1 of 2 RHR heat exchangers
High Pressure Recirculation System (HPRS) - H_1'	Same as H_1 except without the RHR heat exchangers
Low Pressure Recirculation System (LPRS) - H_2	1 of 2 RHR pump trains and 1 of 2 RHR heat exchangers (this was not used in the final analysis!)
Low Pressure Recirculation System (LPRS) - H_2'	Same as H_2 except without the RHR heat exchangers
Intake Cooling Water System (ICWS)	1 of 3 ICW pumps
Component Cooling Water System (CCWS)	1 of 3 CCW pumps and 1 of 2 heat exchangers

2.2.2 S_2-Small LOCA Event Tree

The core status outcomes in Figure B.4 are OK indicating success or no core melt, late core melt (LCM) and early core melt (ECM).

The event tree effectively starts with the success/failure of the AFWS since the RPS (K) is assumed to be successful and PCS (M), i.e., main feedwater, will trip from a safety injection signal and be isolated. If the AFWS (L) is successful, the next consideration is the HPIS (D_1). HPIS success then requires recirculation cooling to achieve adequate core cooling. Given either HPRS (H_1') or LPRS (H_2') success, this leads to an OK core status. Failure of both HPRS (H_1') and LPRS (H_2') leads to a late core melt (Sequence 1). Information indicates that sufficient water from an S_2 LOCA will be channeled to the sump to assure an adequate recirculation water source. Furthermore, the RHR heat exchangers are not required since the AFWS is successful and is adequate to remove all decay heat by itself.

The second major branch of the event tree occurs when the AFWS (L) is successful but the HPIS (D_1) fails. In this case the option that would be selected is the secondary blowdown mode (SBM) labeled X in the event tree. This requires both pressurizer sprays and secondary steam relief by steam bypass to the condenser or opening 2 of 2 atmospheric dump valves. The objective of this mode is to rapidly depressurize the primary so that the LPIS can be brought into service. Subsequent success of the LPIS and the LPRS later in the accident results in an OK sequence. Failure to depressurize leads to an early core melt (Sequence 4). Successful depressurization but failure of LPIS also leads to an early core melt (Sequence 3). Successful depressurization and LPIS but subsequent failure of the LPRS results in a late core melt (Sequence 2).

The third major branch of the S_2 event tree is where the AFWS (L) fails. Since bleed and feed is a viable option in transient situations where the AFWS fails, it is likewise applicable to the small LOCA. In fact, the water lost from the small LOCA may not be enough "bleed" to reduce the PORV (P_1) criteria from 2 of 2 to 1 of 2 PORVs needed to be opened. These PORVs must be remote manually opened by the operator. The HPIS gets a safety injection signal due to the LOCA so an operator initiated HPIS start is not required as it is for a transient bleed and feed.

To continue with the bleed and feed branch of the small LOCA event tree, failure of the HPIS (D_1) leads directly to early core melt (Sequence 7). Given success of D_1, failure to open both PORVs (P_1) also leads to an early core melt (Sequence 6). Finally, even with success of D_1 and P_1, recirculation phase cooling is required with the RHR heat exchangers since the AFWS has already failed. Thus, failure of H_1 after success of D_1 and P_1 leads to a late core melt (Sequence 5).

2.3 Containment Systems Event Tree and Success Criteria

Past PRAs have usually assumed that failure of the containment due to failure of the containment systems to protect against overpressure led to sump flashing and subsequent low pressure (RHR) pump cavitation. At St. Lucie, and perhaps most other PWRs, it is estimated that at most only a foot or so of sump water would flash before the sump water would cool enough to come into equilibrium with the containment environment. Thus, the containment systems can be separated from the core melt event trees and applied in tandem to each accident sequence. In practice it is only necessary to consider the containment event tree for core melt sequence, and further, only to dominant core melt accident sequences. This tends to simplify the analysis process. The containment system event tree is given in Figure B.5.

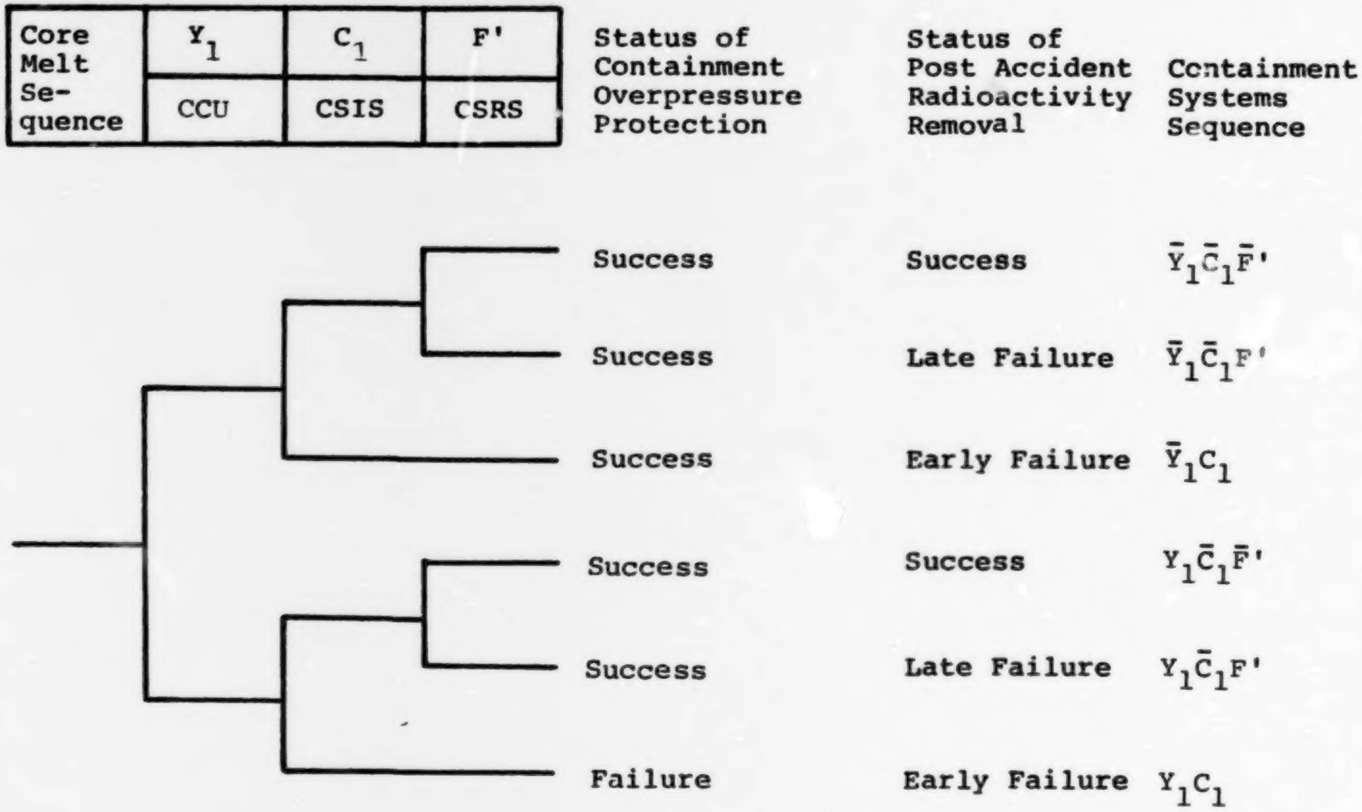
Core Melt Sequence
Y_1
CCU
C_1
CSIS
F'
CSRS
Status of Containment Overpressure Protection
Status of Post Accident Radioactivity Removal
Containment Systems Sequence
Success
Success
$\bar{Y}_1\bar{C}_1\bar{F}'$
Success
Late Failure
$\bar{Y}_1\bar{C}_1F'$
Success
Early Failure
$\bar{Y}_1C_1$
Success
Success
$Y_1\bar{C}_1\bar{F}'$
Success
Late Failure
$Y_1\bar{C}_1F'$
Failure
Early Failure
Y_1C_1

Figure B.5 Containment Systems Event Tree

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2.3.1 Containment Systems Success Criteria

Each heading in the containment systems event tree relates to a system or combination of systems. Each system has a success criteria and a fault tree model. The success criteria are given below.

System	Success Criteria
Containment Cooling Units (CCU) - Y_1	1 of 4 CCUs operating
Containment Spray Injection System (CSIS) - C_1	1 of 2 CSIS pump trains
Containment Spray Recirculation System (CSRS) - F	1 of 2 CSRS pump trains with 1 of 2 RHR heat exchangers

2.3.2 Containment Systems Event Tree

Figure B.5 gives the simplified containment system event tree (CSET) for St. Lucie. This is not to be confused with the traditional containment failure mode event tree which will be discussed later. The purpose of the CSET is to establish the containment state which combined with the core melt accident sequence provide the complete systems accident sequence and the basis for mapping into release categories.

It is not necessary to trace the containment event tree branch by branch since it is symmetrical and the status columns in Figure B.5 are self-explanatory.

In general, what this says is the containment can be protected from overpressure by

1) 1 of 4 operating containment fan units, or

2) 1 of 2 CSIS trains.

Success of post accident radioactivity removal (PARR) requires 1 of 2 early sprays (C_1) and 1 of 2 late sprays (F').

The success as well as failure events are shown in the containment system sequence column. Even the success cases are important since given core melt the containment, by definition, always fails but it is the way it fails and the probability of failure that are important to risk to the public.

3.0 DATA BASE

3.1 Initiating Events and Events Not Modeled

Part of every accident sequence involves events that are not quantified using a fault tree model. These events include the initiating event and certain events such as the failure of the PCS or failure of the pressurizer relief valves to open. The frequency or probability of the combination of non-modeled events in a given accident sequence will be called the multiplier probability. An example would be the accident sequence

$T_3MQX\bar{L}D_1$

where

T_3 is the initiating event,

M, Q, and X are the other non-modeled events, and

$\bar{L}D_1$ is the Boolean accident sequence.

Thus, T_3MQX is the multiplier expression and $p(T_3MQX)$ is the multiplier probability. The non-modeled events are independent so that the individual probabilities can be multiplied together. Continuing the example,

$$p(T_3MQX) = p(T_3) * p(M) * p(Q) * p(X)$$

There are eight events and one pseudo event (the PORV opening when not demanded) which are described below. In general we will use generic data to permit better comparison between plant types and to avoid any controversy regarding the latest or best plant specific data.

Small LOCA (S_2)

$f(S_2)$ = 2.0E-2 per reactor year

This comes from the IREP Procedures Guide NUREG/CR-2728,[3] page 74, Table 5.3-1. It includes the frequency of 0.38" to 1.2" and 1.2" to 1.66" diameter breaks; no distinction is made between these two LOCA sizes in the TAP A-45 analysis. In fact, the S_2 definition used here is a break less than 2" in diameter.

Loss of Offsite Power Transient (T_1)

$f(T_1)$ = 8.6E-2 per reactor year

The frequency quoted in NUREG/CR-2728, Table 5.3-1, is 3.2E-1; however, later generic data is available in the EPRI NSAC/80 July 1984 report.[4] In this report the data through 1983

0251

indicate a frequency of 0.088 per site year, and for the last three years alone (1981-1983) a frequency of 0.027 per site year. While the latter number may seem to be the more reasonable frequency to characterize the current situation, NRC recommendation as stated in NUREG 1032 is that a 0.086 frequency will be used for loss of offsite power.

Transient Initiated by a Total Interruption of Main Feedwater (T_2)

$f(T_2)$ = 1.0E-0 per reactor year

This is the frequency quoted in NUREG/CR-2728, Table 5.3-1, and thus will be considered as generic data.

Transient Initiated by a Loss of a Vital AC (T_4) or DC (T_5) Bus

$f(T_4$ - AC Bus) = 6E-3

$f(T_5$ - DC Bus) = 3E-3

These frequencies are based upon the ANO-1 and ASEP information coupled with our own engineering judgment. This information is summarized below.

AC-Bus Initiating Event

ANO-1	3.5E-2
Original ASEP	6E-2
Revised ASEP	5E-3
Latest ASEP	5E-3
Other Studies	9E-4 to 6E-2

DC-Bus

ANO-1	1.8E-2
Original ASEP	6E-2
Revised ASEP	5E-4
Latest ASEP	6E-3
Other Studies	9E-4 to 6E-2

The magnitude of the ANO frequencies seemed high to us so the ASEP frequency was chosen as the basis. Nevertheless, the factor of 2 ratio between the AC & DC Bus frequencies used in ANO seemed reasonable since there are effectively 2 AC Buses for each DC Bus.

All Other Transients Which Do Not Affect Front Line Systems Significantly (T_3)

$f(T_3)$ = 7.1E-0 per reactor year

As the name implies, this transient covers everything not included in T_1, T_2, T_4, and T_5. Such transients occur frequently. The number quoted in NUREG/CR-2728 Table 5.3-1 is 7.1 occurrences per reactor year.

Failure of Power Conversion System and Secondary Steam Relief (M)

$f(M) = 1.0E-2$ per demand

This is the failure of the PCS mode of removing decay heat given the PCS is initially available following the transient. The 1.0E-2 number has been used in several PRAs starting with Wash 1400 (page V-39) and thus will be considered generic for purposes of this study.

Failure of Depressurization with Secondary Blowdown (X)

$p(X,LOP) = 1.5E-2$ per demand
$p(X,LOP) = 3.8E-2$ per demand

Given an S_2 LOCA or transient induced LOCA and AFWS success, this mode is used to respond to the situation where HPIS fails. If the primary can be depressurized, the LPIS can then provide injection. This depressurization is accomplished using pressurizer sprays and secondary steam relief or dumping beyond that needed with the AFWS. Although secondary blowdown was not modeled in detail, a simplified fault tree was used to establish the failure frequency. The model is shown in Figure B.6. The model is relatively self-explanatory. The Boolean expression using symbols from the fault tree is

$$X = X_1 + (X_2+X_3)(X_4+X_5) + X_6\ [X_7(X_9+X_{10}) + X_8(X_{11}+X_{12})].$$

Assuming these events are independent and the following event failure probabilities,

$p(X_1) = 1.5E-2$ typical operator error failure probability (3E-3 times 5 for stress factor)

$p(X_2) = 1$ for loss of offsite power
$= 6.4E-3$ VCC-LF frequency from generic data table (3E-3 times 2 since it is assumed both valves are needed plus the common mode failure of 2 valves (4E-4))

$p(X_3) = 1$ for loss of offsite power
$= 1E-2$ estimate of pump unavailability

$p(X_4) = 3E-3$ VCC-LF probability from generic data table

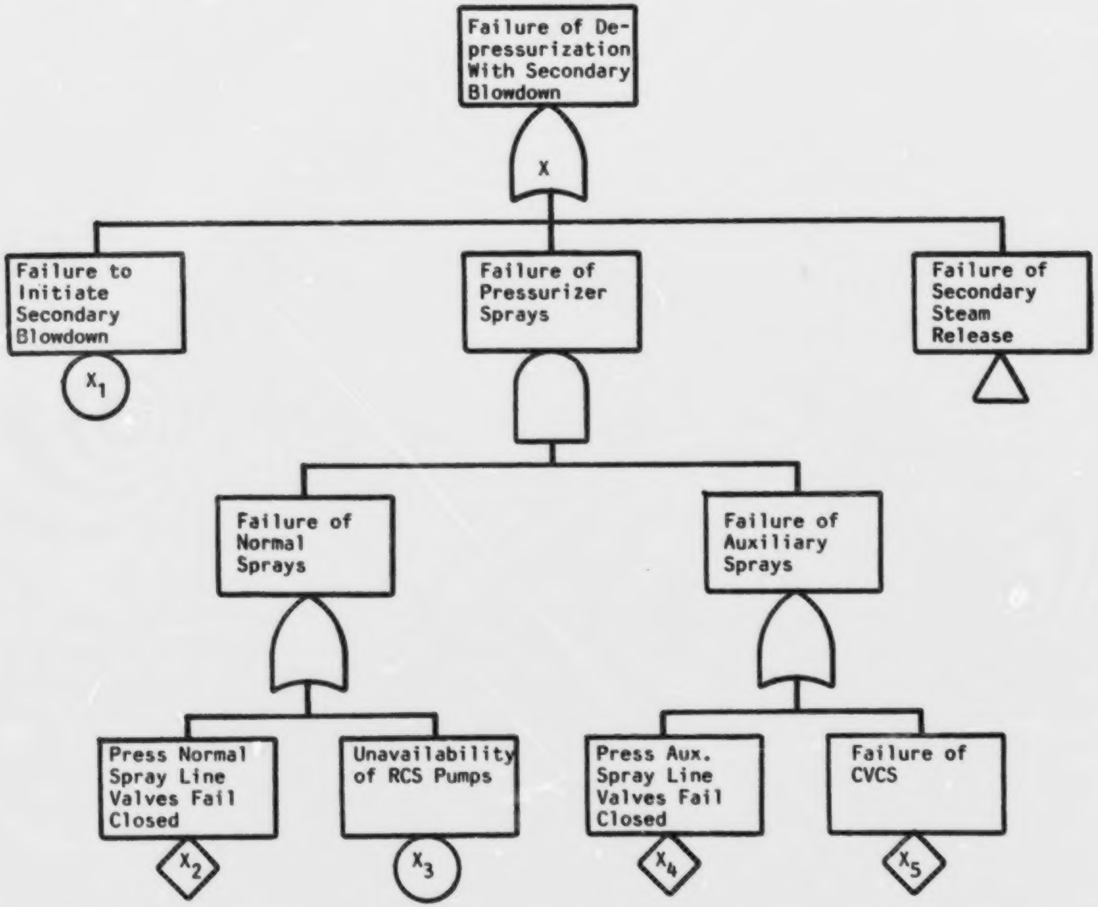
Failure of Depressurization With Secondary Blowdown
X
Failure to Initiate Secondary Blowdown
X_1
Failure of Pressurizer Sprays
Failure of Secondary Steam Release
Failure of Normal Sprays
Failure of Auxiliary Sprays
Press Normal Spray Line Valves Fail Closed
X_2
Unavailability of RCS Pumps
X_3
Press Aux. Spray Line Valves Fail Closed
X_4
Failure of CVCS
X_5

Figure B.6 Simplified Fault Tree for Secondary Blowdown

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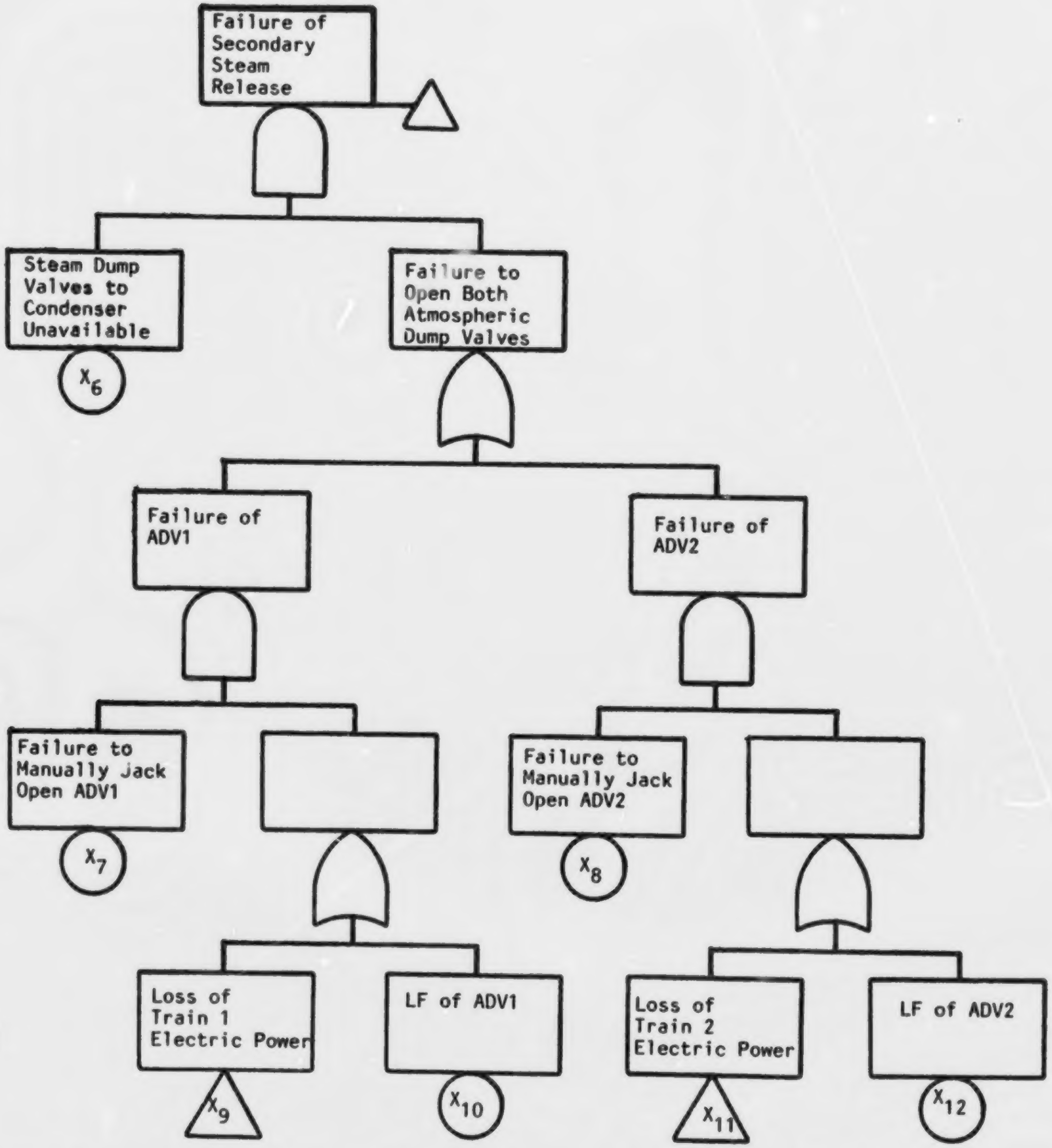
Failure of Secondary Steam Release
Steam Dump Valves to Condenser Unavailable
X_6
Failure to Open Both Atmospheric Dump Valves
Failure of ADV1
Failure of ADV2
Failure to Manually Jack Open ADV1
X_7
Failure to Manually Jack Open ADV2
X_8
Loss of Train 1 Electric Power
X_9
LF of ADV1
X_{10}
Loss of Train 2 Electric Power
X_{11}
LF of ADV2
X_{12}

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Figure B.6 Simplified Fault Tree for Secondary Blowdown (cont.)

$p(X_5)$ = 2E-2 estimate based on typical system failure probabilities

$p(X_6)$ See analysis

$p(X_7) = p(X_8)$ = 1E-2 human failure to be able to manually jack open these valves

$p(X_9) = p(X_{11})$ = 2E-2 failure of an electric power train

$p(X_{10}) = p(X_{12})$ = 3E-4 failure of relief valve to open on demand

The probability of X can be quantified for two cases; loss of offsite power (LOP) for transient induced LOCAs deriving from T_1, and all other initiating events (AOE). Substituting the probabilities from the list above,

$$\begin{aligned} p(X'LOP) &= (1.5E-2) + (1)(3E-3 + 2E-2) \\ &\quad + p(X_6)[(1E-2)(2E-2 + 3E-4) \\ &\quad + (1E-2)(2E-2 + 3E-4)] \\ &= (1.5E-2) + (2.3E-2) + p(X_6)(4E-4) \end{aligned}$$

if we assume $p(X_6) = 1$ to be conservative and since it does not contribute very much,

$$p(X'LOP) = 3.8E-2 \text{ per demand}$$

and,

$$\begin{aligned} p(X'AOE) &= (1.5E-2) + [(6.4E-3) + (1E-2)][(3E-3) \\ &\quad + (2E-2)] + p(X_6)[(1E-2)(2E-2+3E-4) \\ &\quad + (1E-2)(2E-2+3E-4)] \\ &= (1.5E-2) + (3.8E-4) + p(X_6)(4E-4) \end{aligned}$$

Again, if we assume $p(X_6) = 1$,

$$p(X'AOE) = 1.5E-2$$

Probability of the PORVs Opening When Not Needed - In several branches of the event trees the pressurizer relief valves are not required to open to prevent overpressure but, nevertheless, may inadvertently open. Experience has shown that this occurs about 7% of the time. Thus this factor is used in all those cases to account for the likelihood of the relief valves opening when it is subsequently asked whether the valves closed or not, i.e., event Q.

Failure of the Pressurizer Relief Valves to Close Given That They are Open (Q)

$$p(Q) = 3.0E-2$$

We will assume that both PORVs are blocked as the worst case scenario. Given that a pressurizer relief valve is open, the likelihood that it would not close or stick open is 1.0E-2 taken from NUREG/CR-2728, p 127, Table 5.1-1, section 2.7.2. Since there are 3 SRVs the probability is 3 x (1.0E-2) = 3.0E-2.

Failure of the Pressurizer Relief Valves to Open When Demanded (P)

$$p(P) = 1.3E-6$$

There are two cases to be considered given the system has 2 PORVs and 3 SRVs. Each of the PORVs have block valves which may or may not be closed during normal operations. Thus Case 1 is where the plant operates with the block valves open and Case 2 is where the plant operates with the block valves closed. The probabilities are taken from NUREG/CR-2728, page 127, Table 5.1-1 as noted in Table B.2 of this report.

Case 1:

$$\begin{aligned} p(P) &= p(PORV\text{-}CM) * p(SRV\text{-}CM) \\ &= (3.9E-5) * (1.3E-6) = 5.1E-11 \end{aligned}$$

Case 2:

$$p(P) = p(SRV\text{-}CM) = 1.3E-6$$

Case 2 will be used in the base case analysis.

The preceding discussion is the rationale behind the probabilities used to quantify the multipliers for the accident sequences evaluated in the St. Lucie internal analysis. Table B.1 shows the multiplier frequency for each non-Boolean Multiplier expression established by the event trees. The complement events ($\bar{X}$ and $\bar{Q}$) were also considered.

3.2 Basic Event Probabilities

The data used for the quantification of the St. Lucie fault trees was collected from a variety of references. Generally, the IREP procedures guide was the source for most hardware failure rates, the Station Blackout Accident Analysis (TAP-A44) was used for most electrical failure rates and the Human Factors Handbook for most failures due to human errors.

The data in Table B.2 was used in the quantification of the St. Lucie fault trees. The table lists the event ID, a brief description of the failure, the failure rate and the reference.

This is generic data and was used to generate the SETS value block in which probabilities are assigned to every basic event in the fault tree. For example, the event HPRMV38-VCC-LF is

0257

Table B.1 Frequency of Initiating Events and Event Tree Headings Not Modeled

	S_2	T_1	T_2	T_3	M	X	P	Freq.PORVs Open When Not Needed	Q	Multiplier Frequency
S_2M	2E-2				1E-0					2.0E-2
$S_2M\bar{X}$	2E-2				1E-0	.985				2.0E-2
S_2MX	2E-2				1E-0	1.5E-2				3.0E-4
$T_1M\bar{P}\bar{Q}$		8.6E-2			1E-0				.97	8.3E-2
T_1MP		8.6E-2			1E-0		1.3E-6			1.1E-7
T_1MQ		8.6E-2			1E-0			7E-2	3E-2	1.8E-4
$T_1MQ\bar{X}$		8.6E-2			1E-0	.965		7E-2	3E-2	1.7E-4
T_1MQX		8.6E-2			1E-0	3.8E-2		7E-2	3E-2	6.9E-6
$T_1M\bar{P}Q$		8.6E-2			1E-0				3E-2	2.6E-3
$T_2M\bar{P}\bar{Q}$			1E-0		1E-0				.97	9.7E-1
T_2MP			1E-0		1E-0		1.3E-6			1.3E-6
T_2MQ			1E-0		1E-0			7E-2	3E-2	2.1E-3
$T_2MQ\bar{X}$			1E-0		1E-0	.985		7E-2	3E-2	2.1E-3
T_2MQX			1E-0		1E-0	1.5E-2		7E-2	3E-2	3.2E-5
$T_2M\bar{P}Q$			1E-0		1E-0				3E-2	3.0E-2
$T_3M\bar{P}\bar{Q}$				7.1E-0	1E-2				.97	6.9E-2
T_3MP				7.1E-0	1E-2		1.3E-6			9.2E-8
T_3MQ				7.1E-0	1E-2			7E-2	3E-2	1.5E-4
$T_3MQ\bar{X}$				7.1E-0	1E-2	.985		7E-2	3E-2	1.5E-4
T_3MQX				7.1E-0	1E-2	1.5E-2		7E-2	3E-2	2.2E-6
$T_3M\bar{P}Q$				7.1E-0	1E-2				3E-2	2.1E-3
T_3Q				7.1E-0				7E-2	3E-2	1.5E-2
$T_3Q\bar{X}$				7.1E-0		.985		7E-2	3E-2	1.5E-2
T_3QX				7.1E-0		1.5E-2		7E-2	3E-2	2.2E-4

0258

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Table B.1 Frequency of Initiating Events and Event Tree Headings Not Modeled (cont.)

	T_4	T_5	M	X	P	Frequency PORVs Open When Not Needed	Q	Multiplier Frequency
$T_4M\bar{P}\bar{Q}$	6E-3		1E-0				.97	5.8E-3
T_4MP	6E-3		1E-0		1.3E-6			7.8E-9
T_4MQ	6E-3		1E-0			7E-2	3E-2	1.3E-5
$T_4MQ\bar{X}$	6E-3		1E-0	.985		7E-2	3E-2	1.2E-5
T_4MQX	6E-3		1E-0	1.5E-2		7E-2	3E-2	1.9E-7
$T_4M\bar{P}Q$	6E-3		1E-0				3E-2	1.8E-4
$T_5M\bar{P}\bar{Q}$		3E-3	1E-0				.97	2.9E-3
T_5MP		3E-3	1E-0		1.3E-6			3.9E-9
T_5MQ		3E-3	1E-0			7E-2	3E-2	6.3E-6
$T_5MQ\bar{X}$		3E-3	1E-0	.985		7E-2	3E-2	6.2E-6
T_5MQX		3E-3	1E-0	1.5E-2		7E-2	3E-2	9.5E-8
$T_5M\bar{P}Q$		3E-3	1E-0				3E-2	9.0E-5

Table B.2 St. Lucie Failure Event Data

Fault ID	Fault Description	Failure Prob.	Source
VCC-LF	MOV, NCFC (local faults)	8E-3	Ref. 1, Item 2.1.1 (3E-3) plus control circuit failures (5E-3)
VOO-LF	MOV, NOFO	8E-3	As above
VOC-LF	MOV, NOFC	3.6E-5	Ref. 1, Item 2.1.2 (1/2) (1E-7/hr) (720/hrs) monthly test interval
XOC-LF	Manual Valve, NOFC	1E-4	Ref. 1, Item 2.6 2 (1/2) (3E-7/hr) (720 hrs) monthly test interval
CCC-LF	Check Valve, NCFC	1E-4	Ref. 3, Item 2.4.1
NCC-LF	Pneumatic-Hyd. Valve, NCFC	8E-3	Ref. 1, Item 2.3.1 (3E-3) plus control circuit failures (5E-3)
BLKVLV-LF	PORV Block Valves, NCFC	8E-3	Ref. 1, Item 2.1.1 (3E-3) plus control circuit failures (5E-3)
PORV-LF	PORV Fails to Open, NCFC	3E-4	Ref. 1, Item 2.9.1
CB-LF	Circuit Breaker	3E-5	Ref. 1, Item 4.1.2
Fan-LF	Cooling Fan	2.4E-4	Ref. 1, Item 4.17.1, normally operating, 1E-5/hr, assume accident duration 24 hrs.
BUS-416-1A2-LF BUS-416-1B2-LF	LF of 4KV bus plus 2 CB and 1 transformer	8.4E-5	Ref. 1, Items 4.3.1, 4.1.2 and 4.5.1 (1E-8*24 + 6E-5 + 2.4E-5)
BUS-416-1A3-LF BUS-416-1B3-LF	LF of 4KV bus plus 1 CB	3.0E-5	Ref. 1, Items 4.3.1 and 4.1.2 (1E-8*24 hr + 3E-5)
BUS-416-1AB-LF	LF of 4KV bus plus 2 CBs	6.0E-5	Ref. 1, Items 4.3.1 and 4.1.2 (1E-8*24 hr + 2*3E-5)
BUS-480-1B2-LF BUS-480-1A2-LF	LF of 480 AC bus plus 2 CBs and 1 transfoormer	8.4E-5	Ref. 1, Items 4.3.1, 4.1.2 and 4.5.1 (1E-8*24 + 2*3E-5 + 2.4E-5)

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Table B.2 St. Lucie Failure Event Data (cont.)

Fault ID	Fault Description	Failure Prob.	Source
MCC-480-1A5-LF MCC-480-1A6-LF MCC-480-1B6-LF MCC-480-1B5-LF	LF of 480 AC MCC plus 1 CB	3.0E-5	Ref. 1, Items 4.3.1 and 4.1.2 (1E-8*24 + 3E-5)
BUS-125-1A-LF BUS-125-1B-LF	LF of 125V DC bus plus 2 CB	6.0E-5	Ref. 1, Items 4.3.1 and 4.1.2 (1E-8*24 hr + 2*3E-5)
ACT-LF	Actuation local fault	5E-3	Ref. 4
BATTCHG-LF	Local faults of battery charger plus CB failure	5.4E-5	Ref. 1, Item 4.10.1 (1E-6/hr*24 hr) plus Item 4.1.2 (3E-5)
BAT-LF	Battery local faults	1.8E-3	See text
DG-LF	Diesel gen. local faults	3.8E-2	See test
PMS-LF	Motor pump NSFS	8.7E-3	FTS (3E-3) Ref. 1, Item 1.1.1 plus FTR (3E-5/hr*24 hr) Ref. 1, Item 1.1.2.1 plus control ckt failures (5E-3)
PTS-LF	Turbine pump NSFS	3.5E-2	FTS (3E-2) Ref. 1, Item 1.2.1 plus FTR (1E-5/hr * 24 hr), Ref. 1, Item 1.2.2 plus control ckt failures (5E-3)
LOSS-OSP	Non-initiator loss of offsite power	1E-3	Typical number used in past PRAs
HX-FB	Heat exchanger flow blockage	3E-4	Ref. 1, Item 4.4.1 Used # for orifice plug
OPF-MACT-FNB	Op. fails to manually initiate Feed & Bleed	3E-3	Ref. 2, Table 20-7, Item 3 short list, no checkoff
VOC-UTM	MOV not restored after T or M or downtime for T or M	3.8E-4	Ref. 2, Table 20-7, Item 1 fail to restore to proper position after T or M. (1E-3)(1/12 monthly check) plus 3E-4 due to downtime
XOC-UTM	Manual valve T or M	3.8E-4	As above

Table B.2 St. Lucie Failure Event Data (cont.)

Fault ID	Fault Description	Failure Prob.	Source
VCC-UTM		3E-4	Receives signal to open but could be down for T or M
NCC-UTM		3.0E-4	As above
BAT-UTM	T or M faults of batteries	1.1E-3	Ref. 2, Table 20-7, Item 1 (1E-3)(1/12) plus 1E-3 due to downtime
DG-UTM	Diesel Generator T or M faults	6.1E-3	Ref. 2, Table 20-7, Item 1 (1E-3)(1/12) plus 6E-3 due to downtime
PMS-UTM	Motor Pump T or M faults	2.1E-3	Ref. 2, Table 20-7, Item 1 fail to restore to proper position after T or M. (1E-3)(1/12 monthly check) plus 2E-3 due to downtime
PTS-UTM	Turbine Pump T or M faults	6.1E-3	See text
ICW-PMS-UTM	Intake cooling pump T or M faults	1E-2	Ref. 2, Table 20-7, Item 1, fail to restore pump after T&M (1E-3)*(1/12 monthly check) plus 1E-2 downtime - standby pumps
HX-UTM	Heat exchanger T or M faults	6E-4	Due to downtime
Valve-CM	Valve common mode	4E-4	See text
Battery-CM	Battery common mode	4.4E-4	See text
Diesel-CM	Diesel generator common mode	1.5E-3	See text
Pump-CM	Motor pump common mode		See text
PORV-CM	PORV common mode	3.9E-5	See text
BLKVLV-CM	Block valve common mode	4E-4	See text
Sens-CM	SIS sensor group common mode failure	3E-5	Ref. 4

0262

Table B.2 St. Lucie Failure Event Data (cont.)

Fault ID	Fault Description	Failure Prob.	Source
LOG-UTM-CM	T or M common mode failure of both logic trains	2E-4	1/10 Beta factor (2E-3) logic train T or M failures, Ref. 4

0263

given a frequency of 8E-3 found in Table B.2 under VCC-LF which is the same frequency for any motor operated valve that is normally closed and fails to open on demand, i.e., fails closed. Thus the table represents many more events than are actually shown. Some specific events are given in Table B.2 as necessary.

Some failures are listed as having more than one failure rate. This results from the effects of the loss of offsite power (LOP) as an initiating event on the other components in the plant, specifically the component cooling water and service water pumps which are normally running. Since these pumps are running during power operations the non-LOP failure rate consists of a failure to keep running. However, given loss of offsite power, these pumps stop and the failure rate becomes the sum of a failure to start and a failure to continue running.

3.3 Failure Probability Rationale

Several event probabilities deserve further exploration than can be conveniently placed in the left hand column of Table B.2.

3.3.1 Battery Failure Rate

St. Lucie conducts a monthly individual cell voltage and specific gravity test. Such tests, according to NUREG-0666, will detect many of the potential problems that could lead to battery failure, but not all the problems. A partial discharge test would probably be sufficient to detect most of the potential failures not found during the quarterly test. The discharge tests conducted at St. Lucie are done on an 18 month interval. In discussions with the TAP A-30 (DC power) people it was estimated that 75% to 95% of all the possible battery failure modes can be detected by the quarterly tests. The remaining faults require a discharge test for detection. Therefore the battery local fault probability is based on 85% of the faults being detected with monthly tests and 15% by the discharge test which leads to:

$$\text{BAT-LF} = 1/2 * (0.85 * 9.2\text{E-}7/\text{h}) * 2190 \text{ hours} + 1/2 * (0.15 * 9.2\text{E-}7/\text{h}) * 8760 \text{ hours/year} * 1.5 \text{ years} = 8.56\text{E-}4 + 9.07\text{E-}4 = \underline{1.8\text{E-}3}$$

3.3.2 Local Control Circuit Failures

There is considerable variation in the potential faults applicable to any component and in the faults which are included in the failure probability data given in different data bases. This is especially so for pumps and motor operated valves. We have typically segregated component failure into local faults (LF) and unavailability due to test and maintenance (UTM). The LF normally include the failure to start or operate on demand and the failure to continue running or remain open. For pumps it includes the pump and driver. For valves it includes the valve and its driver. It is not always clear if the interfacing control circuits are included. Therefore, we propose the following, starting with the NUREG/CR-2728 generic data base (Table 5.1-1).

MDP-LF = 3E-3/d(FTS) + 3E-5/h(FTR) * 24h + 5E-3(cc) = 8.7E-3
TDP-LF = 3E-2/d(FTS) + 1E-5/h(FTR) * 24h + 5E-3(cc) = 3.5E-2
DDP-LF = 1E-3/d(FTS) + 8E-4/h(FTR) * 24h + 5E-3(cc) = 2.5E-2

where the control circuit (cc) failure probability is derived as follows for the MDP:

Failure of the control circuit breaker to transfer Table 5.1-1 Item 4.1.1	3E-3
Faults in the control circuit logic ($1/2\lambda T$) 1/2 * 3E-6 * 720h Table 5.1-1 item 4.14	1E-3
Open circuit in Cable ($1/2\lambda T$) 1/2 * 3E-6 * 720h Table 5.1-1 item 4.13.1	1E-3
	5E-3

This obviously is not a detailed analysis of the hardware. Furthermore, we have added some control circuit failure probability to all three pump types even though we have some verbal opinions stating that this probability could be higher or lower for the TDP and DDP. Unfortunately, arriving at failure probabilities from past data bases is not a precise science (even though it should be). These numbers compare well to the corresponding numbers derived from the NUREG/CR-3226 (A-44), Table E-1 page 228 which includes control circuits as part of the demand (d) failure, i.e.,

MDP-LF 2E-3/d(FTS) + 2E-5/h(FTR) x 24 = 2.5E-3
TDP-LF 2E-2/d(FTS) + 5E-5/h(FTR) x 24 = 2.1E-2
DDP-LF 2E-2/d(FTS) + 1E-4/h(FTR) x 24 = 2.2E-2

Similarly for electrically operated valves (e.g., MOV, SOV) that must change positions, the 5E-3 control circuit failure probability would need to be added to the hardware failures already considered to obtain the total LF probability.

3.3.3 Unavailability Due to T&M Downtime

It could be assumed that downtime is negligible since routine maintenance is not performed during operation and failures requiring maintenance or repair were relatively infrequent. Industry wide data seems to be contrary to this, so we will use downtime unavailabilities representative of the data seen in several studies. This unavailability will be added to the failure to restore probability to obtain the total UTM probability. The downtime probabilities to be used are:

0265

All valves (except check valves) regardless of position	3E-4
Motor driven pump SW & CCW	1E-2
Motor driven pump, other	2E-3
Turbine driven pump	6E-3
Diesel driven pump	1E-2
Heat exchanger	6E-4
Containment fans	5E-3
Battery	1E-3
Diesel generator	6E-3

3.3.4 Common Mode Failures

Diesel Generator

The following is based on information found in NUREG/CR-1032,[9] pp 23-25:

$$\text{DG-FTS} = \frac{\text{272 Failures}}{\text{14304 Demands}} = 1.9\text{E-}2$$

$$\text{DG-FTR} = (2.4\text{E-}3/\text{h}) * 8\text{h} = 1.9\text{E-}2$$

$$\begin{aligned}\text{DG-LF} &= \text{DG-FTS} + \text{DG-FTR}\\ &= 1.9\text{E-}2 + 1.9\text{E-}2 = \underline{3.8\text{E-}2}\end{aligned}$$

$$\text{DG-CM factor} = \frac{\text{11 Identified CM Failures}}{\text{272 DG Failures}} = 4.0\text{E-}2$$

$$\begin{aligned}\text{DG-CM} &= \text{DG-LF} * \text{DG-CM factor}\\ &= 3.8\text{E-}2 * 4.0\text{E-}2 = \underline{1.5\text{E-}3}\end{aligned}$$

Batteries

The following calculations are based on information found in NUREG/CR-0666[10] (TAP A-30):

8 failures in 996 battery years (page E-6)

$$\lambda = \frac{\text{8 Failures}}{\text{996 Years} * \text{8760 hr/yr}} = 9.2\text{E-}7/\text{h}$$

BAT-LF = 3.9E-3 based on the 9.2E-7/h failure rate as discussed previously in the letter. For common mode consider (page E-7):

1 common mode occurrence in 996 battery years or 498 battery system years (2 batteries/system)

$$\lambda_{CM} = \frac{\text{1 CM Failure}}{\text{498 BAT SYS YRS}} = 2.0E\text{-}3/yr$$

Using the same approach used for the LF calculation

BAT-CM = 1/2 * (0.85 * 2.0E-3/yr) x 1/4 yr +

1/2 * (0.15 * 2.0E-3/yr) x 1.5 yr =

2.13E-4 + 2.25E-4 = 4.4E-4

Pumps

One approach to the common mode failure of the three AFWS pumps at St. Lucie (1 turbine and 2 motor) uses a 0.06 β factor derived from the Midland and LaSalle PRAs and an "average" pump failure probability where the pump failures include failure to start, failure to continue operating and interfacing control circuits for the motor driven pumps. Using a 3.5E-2 for the TDP and 8.7E-3 for the MDP, the common mode probability is

$$(6E\text{-}2) * [(3.5E\text{-}2) * (8.7E\text{-}3)^2]^{1/3} = 8E\text{-}4$$

If the interfacing control circuit probabilities were excluded, this calculation would become

$$(6E\text{-}2) * [(3E\text{-}2)(3.7E\text{-}3)^2]^{1/3} = 4.5E\text{-}4$$

The purpose of the above calculations is an attempt to derive the common mode probability from a β factor currently in use and the basic pump failure probabilities. It has been suggested that the ASEP information from the March 21, 1985, memo to the ASEP Senior Consultant Group is more accurate. That is, for AFWS:

2 trains (1MDP & 1TDP) 4E-4 non-staggered testing

2 trains (2TD) 5E-4 non-staggered testing

3 trains (mix of pumps - not specified) 2E-4 non-staggered testing

The common mode between 2 motor driven pumps could be estimated using the 6E-2 β factor, i.e., (6E-2)(8.7E-3) = 5.2E-4. One would expect that adding a third pump of a different type should decrease the common mode probability. Thus the 2E-4 probability seems reasonable and we will use it for non-staggered testing.

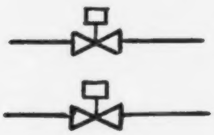
The St. Lucie intake cooling water has two pumps normally operating (but which may have to be restarted in some cases) and one pump on standby. The common mode failure probability to be used is 1E-5 based on available data for staggered testing of service water pumps. Two of the three component cooling water pumps are also normally running with the third one on standby. The available data for common mode failure of CCW pumps which are staggered testing indicates that 1E-5 is a best estimate.

Common mode for other pumps (HPI, LPI, CSI) will be based on a compromise between numbers derived from a 0.06 β factor times the 8.7E-3 motor driven pump failure probability which is 5E-4, and the ASEP numbers. 3E-5 will be used for the three HPI pumps, while 7E-5 will be used for the two LPI pumps and two CSI pumps. All of these pumps are also tested on a staggered basis.

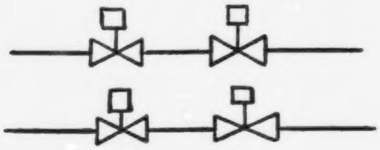
Motor Operated Valves

Common mode on all pairs of motor operated valves that are normally closed and need to be operated/opened for a system to function will be considered. Manual valve and check valve common mode will not be considered. The local fault failure probability for a MOV is the sum of the failure to open on demand (3E-3) plus the failure of interfacing control circuits (5E-3) which sums to 8E-3. Using a 0.05 beta factor from LaSalle and Midland (actually 0.042) the common mode MOV failure probability is 4E-4.

The 4E-4 common mode failure probability for MOVs will be applied to pairs of MOVs and special cases of 4 MOVs as shown below:



or



We believe that the common mode failure probability for 3 or 4 valves should be slightly lower since not all common mode influences would be independent of the number of valves. Our engineering judgment leads us to propose 1.6E-4 for 3 valves and 1.0E-4 for 4 or more valves. Examples of 3 valves are:

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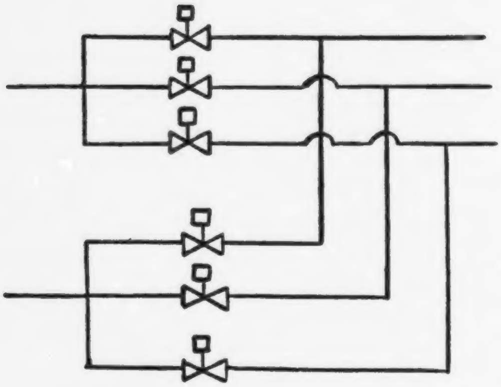
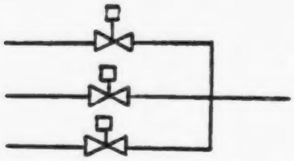
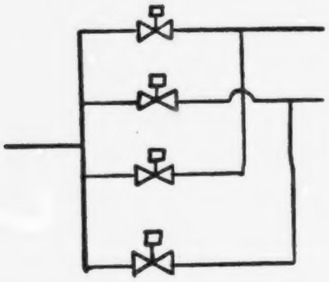


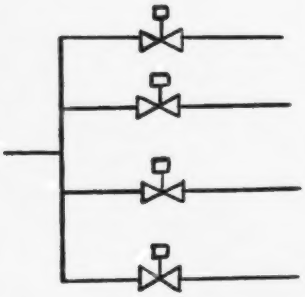
or



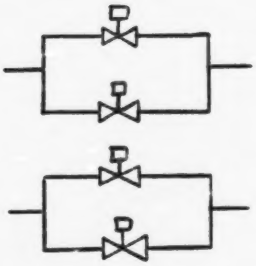
Examples of 4 valves where the common mode failure probability would be reduced are:



or



or



Each situation encountered will need to be evaluated according to its function.

The common mode failure of a "relief" valve due to common mode is derived from a 0.13 beta factor from the LaSalle screening estimates leading to the following common mode failure probabilities:

Pressurizer SRV FTO LF	1E-5	Common Mode	1.3E-6
Pressurizer PORV FTO or FTOP LF	3E-4	Common Mode	3.9E-5
Secondary ADV FTO or FTOP LF	3E-4	Common Mode	3.9E-5

4.0 RECOVERY ANALYSIS

The analysis for the application of operator recovery on the Dominant Accident Sequences (DAS) calculated for St. Lucie is a four step process.

First, the time available for recovery must be determined. This requires an estimate of the time interval between the initiating event and the point at which the core begins to uncover (i.e., the point of no return). This recovery time is based on the work performed in the Accident Sequence Evaluation Program (ASEP) which, in turn, utilized the accident analysis

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in the Generic Feedwater Studies. However, since the ASEP work was concerned with a limited number of sequences the results were extrapolated in order to cover all the dominant sequences analyzed for St. Lucie (see Table B.3). In some specific cases a particular cut set may extend the amount of time available for recovery beyond that determined by the thermal hydraulic analysis. For example, a pump fails due to the absence of room cooling which may take 30 minutes. This extends the recovery time since the actual accident sequence does not start until the pump has failed.

The second step in the recovery process is to identify those failures or basic events which are, in fact, recoverable. The potentially recoverable failures are examined individually to determine if recovery is possible and by what method it can be accomplished. An entire range of events are considered (see Table B.4) from recovery of offsite power to manually actuating safety injection on a failure of the automatic actuation system. Some events which may at first appear to be recoverable, in reality are not. For example, if a valve on the suction of an SI pump fails to automatically open, the pump could cavitate and burn up. Manually opening the valve would not recover the cut set because of the permanent failure of the pump which subsequently occurred.

The third step is to quantify the recovery actions by determining the non-recovery probabilities associated with each action. It is the failure to recover probability which is multiplied with the cut set frequency to determine the reduction in the contributions to core melt. The non-recovery probabilities for electrical failures are those listed in the Station Blackout Accident Analyses NUREG/CR-3226[6] and were expanded in the Accident Sequence Evaluation Program. The non-recovery probabilities for all other failures are based on the model used in the ANO-1 IREP analysis (Ref. 3).

The final step in the recovery analysis is to re-quantify the DAS. This is performed by computer using the accident sequence in equation form with non-recovery factors included. The Boolean equation is entered into the SETS computer code and new core melt probabilities are calculated.

In the actual application of the recovery methodology, a number of assumptions were made about the operating status of the plant and the feasibility of specific recovery actions. These assumptions are listed below.

1) Hardware failures are not directly recoverable, i.e., local faults of pumps or valves cannot be repaired with the exception of control circuit failures of, or loss of power to, motor operated valves.

Table B.3 St. Lucie Dominant Accident Sequence Recovery Timing

Sequence	Time Available for Recovery	Source
$S_2MH_1'H_2'$	2-4 hrs (H)	S_2MD_{12} plus 3 hours for successful injection
$T_3QH_1H_2'$	"	
$T_2MQH_1'H_2'$		
$S_2MD_1D_2$	30-40 min (D)	ASEP S_2D
$T_1MQD_1D_2$	60-70 min (F)	ASEP TMQD (Q=SRV failure)
$T_2MQD_1D_2$	"	"
$T_3MQD_1D_2$	"	"
$T_3QD_1D_2$	"	"
$T_4MQD_1D_2$	"	"
S_2MXD_1	30-40 min (D)	ASEP S_2D
T_1MQLD_1	60-70 min (F)	ASEP TMQD (Q=SRV failure)
T_3QXD_1		
T_1MLE	70-120 min (G)	ASEP TML
T_2MLE	"	"
T_3MLE	"	"
T_4MLE	"	"
T_5MLE	"	"
T_2MLH_1	2-4 hrs (H)	ASEP TML Plus 3 hours
T_5MQLP_1	70-120 min (G)	ASEP TML
S_2MD_1	30-40 min (D)	ASEP S_2D

0271

Table B.3 St. Lucie Dominant Accident Sequence Recovery Timing (cont.)

Sequence	Time Available for Recovery	Source
T_1MQD_1	60-70 min (F)	ASEP TMQD
T_2MQD_1	"	"
T_3MQD_1	"	"
T_4QD_1	"	"
T_3QD_1	"	"
S_2ML	60-70 min (F)	ASEP TMQD
T_1MQL	"	"
T_2MQL	"	"
T_3MQL	"	"
T_4MQL	"	"
T_5MQL	"	"
T_3QL	"	"
T_1ML	70-120 min (G)	ASEP TML
T_2ML	"	"
T_3ML	"	"
T_4ML	"	"
T_5ML		"

0272

Table B.4 St. Lucie Non-Recovery Probabilities

Recovery of:		A 5-10 min.	B 10-20 min.	C 20-30 min.	D 30-40 min.	E 40-60 min.	F 60-70 min.	G 70-120 min.	H 2-4 hrs.
LOSP	RA-1	0.7	0.7	0.5	0.4	0.3	0.3	0.2	0.1
PCS	RA-2	1.0	1.0	1.0	0.9	0.6	0.1	0.03	0.01
Auto-Actuation Failure (from Control Room)	RA-6	0.3	0.1	0.05	0.03	0.03	0.01	0.01	0.01
Electrically realign buses (from control room)	RA-7	0.3	0.1	0.05	0.03	0.03	0.01	0.01	0.01
Battery CM	RA-8	1.0	1.0	1.0	1.0	1.0	1.0	0.9	0.8
Battery LF	RA-9	1.0	1.0	0.9	0.8	0.7	0.7	0.5	0.2
DG-CM	RA-10	1.0	1.0	1.0	1.0	0.9	0.9	0.9	0.7
DG-LF	RA-11	1.0	1.0	1.0	1.0	0.9	0.9	0.9	0.8
Isolation Valve Failure (locally)	RA-14	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01
ICW Discharge Valve or Pump Failure (locally open alternate path)	RA-16	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01
AFW Turbine Steam Supply Failure (locally realign)	RA-17	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01
CCW Pump Failure (start idle CCW pump from control room)	RA-18	0.3	0.1	0.05	0.03	0.03	0.01	0.01	0.01
HPI Discharge Valves Fail to Auto Open (locally open valves manually)	RA-19	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01

Table B.4 St. Lucie Non-Recovery Probabilities (cont.)

Recovery of:		A 5-10 min.	B 10-20 min.	C 20-30 min.	D 30-40 min.	E 40-60 min.	F 60-70 min.	G 70-120 min.	H 2-4 hrs.
CCW Discharge Valve Failure (locally open alternate path)	RA-20	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01
Low Pressure Flow Blockage	RA-24	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01
AFW Feedwater Valve Failure	RA-25	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01
CCW Suction Valve Failure	RA-26	1.0	0.3	0.1	0.05	0.03	0.03	0.01	0.01
Manually open MOV	RA-VCC	1.0	0.56	0.44	0.41	0.39	0.39	0.38	0.38

2) Diesel generator jacket cooling and pump cooling failures extend the time available for recovery, but are not, themselves, recoverable.

3) Pump cavitation faults are not recoverable.

4) Power conversion system (PCS) is recoverable only if offsite power and service water are available.

5) The Q failures (stuck open pressurizer relief valve) are usually considered to be a failure of a PORV to reclose (which is recoverable by closing the block valve) since the PORV pressure setpoints are approximately 100 psi lower than those for SRVs. However, the block valves for St. Lucie are assumed to be normally closed so Q is a stuck open SRV which is not recoverable.

6) Normally, credit is given for the operators to perform a single recovery action for each cut set. The only exceptions are: a) for the recovery of a loss of offsite power which is assumed to be attempted by someone other than the control room operators, b) for the recovery of the PCS from the control room, and c) for sequences in the 2-4 hour time frame where two recovery actions are permitted.

7) Successful safety injection was assumed to extend accident sequence timing by approximately three hours over safety injection failure sequences.

5.0 ACCIDENT SEQUENCE ANALYSIS

Once the system success criteria were established and the event trees constructed, the fault tree construction addressed each system or variation on a system that appears in the event tree headings. The data base for quantification was then established for every independent event found in the event trees or fault trees including the containment systems that are part of the separate containment systems event tree. The reason for the separate core melt and containment system event trees is that the potential modifications were based on core melt accident sequences alone, but the value-impact analysis considers risk to the public which involves containment system performance and containment failure modes.

The initial plant visit was to collect information for special emergencies and to discuss numerous internal analysis concerns such as system success criteria, emergency operating procedures, safety system configurations, and in general any significant changes not documented in the FSAR. This plant visit could have come at any time from near the beginning before the fault trees are started to as late as after the DAS have been established and recovery analyses have been completed. In this case the preliminary event trees, fault

0275

trees, and data base were completed before the initial plant visit.

The trip report included a long list of questions and answers, and observations which was later communicated to the utility for comments. While this is not part of the appendix, it did provide a means to verify a number of points that are used throughout this PRA.

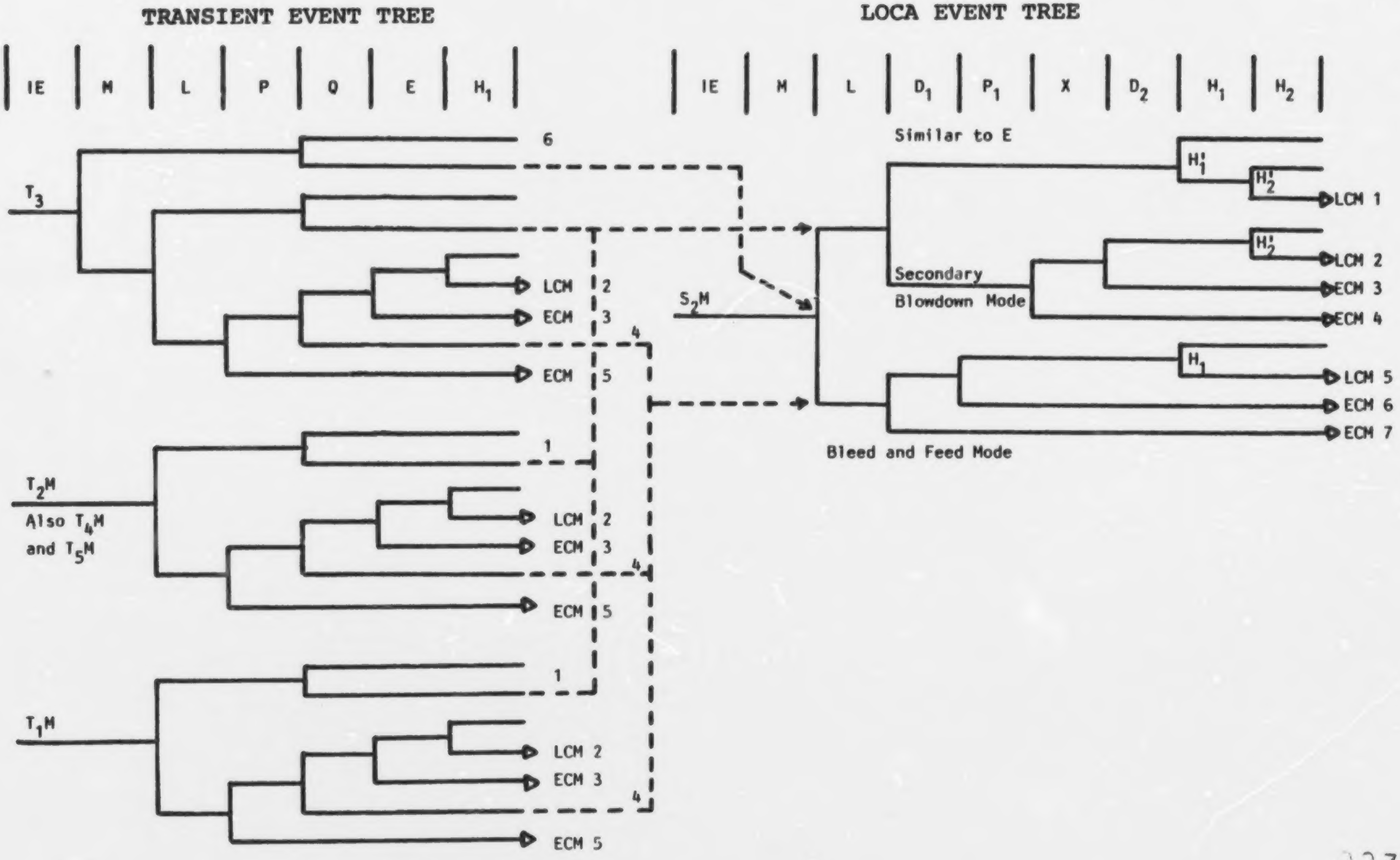
5.1 Accident Sequence Delineation

The first step in the quantification is the delineation of accident sequences. Each LOCA, transient or transient-induced LOCA accident sequence, is identified by a group of symbols representing events that together result in a possible core melt. A condensed version of the event trees found in Sections 2.1 and 2.2 is given in Figure B.7 which shows the connections between the transient and LOCA event trees. The core melt sequences for each initiating event are numbered and designated as late core melt (LCM) or early core melt (ECM). One can trace through the branches of any event tree from the initiating event to end result of success or core melt. The path delineates the success and failure of events (systems and functions) leading to core melt. (Note that we are not interested in the success sequences.) A list of definitions is given in Table B.5.

In this list of definitions, failure of 1 of 4 CCUs is defined as the failure to successfully cool the containment using 1 out of 4 containment cooling fans. The failure of this success criteria would require the loss of all four fans. Therefore, the definition should be read as "Failure of the successful operation of 1 of 4 CCUs."

An example of a path from a transient event tree to the LOCA event tree leading to core melt would be $T_3ML\bar{P}Q\bar{D}_1\bar{P}_1H_1$ i.e., transient event T_3 Sequence Number 4 transferring to LOCA Sequence Number 5. The bar over the symbols means success. In words, this is a transient with offsite power available and the power conversion system initially available (T_3) with subsequent failure of the Power Conversion System (M) and the Auxiliary Feedwater System (L) resulting in the safety relief valves opening successfully (P) but failing to close (Q) leading to a transient induced LOCA. This LOCA starts on the LOCA tree where the Auxiliary Feedwater System (L) has failed and follows the path where the High Pressure Injection System (D_1) and Power Operated Relief Valves (P_1) succeed, that is successful bleed and feed, but leading to a late core melt due to eventual failure of makeup and cooling in the recirculation mode (HPRS with RHR heat exchangers) event H_1.

Typically, an accident sequence is stated in terms of its failures which, for the example described above, would be


TRANSIENT EVENT TREE
IE
M
L
P
Q
E
H_1
T_3
6
LCM 2
ECM 3
4
ECM 5
T_2M
Also T_4M
and T_5M
1
LCM 2
ECM 3
4
ECM 5
T_1M
1
LCM 2
ECM 3
4
ECM 5
LOCA EVENT TREE
IE
M
L
D_1
P_1
X
D_2
H_1
H_2
Similar to E
S_2M
H_1'
H_2'
LCM 1
H_2'
LCM 2
Secondary
Blowdown Mode
ECM 3
ECM 4
H_1
LCM 5
ECM 6
ECM 7
Bleed and Feed Mode

0277

Figure B.7 St. Lucie SDHR System Event Trees

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Table B.5 St. Lucie Event Tree/ Fault Tree Definitions

T_1	Loss of offsite power transient
T_2	Loss of power conversion system transient
T_3	Transient with PCS initially available
S_2	Small LOCA (< 2" diameter break)
M	Failure of 1 of 2 PCS trains
L	Failure of 1 of 3 AFWS trains
P	Safety or relief valves fail to open
Q	Safety or relief valves fail to close
E	Failure of bleed and feed mode
P_1	Failure to open 2 of 2 PORVs
D_1	Failure of 1 of 3 HPIS trains
D_2	Failure of 1 of 2 LPIS trains
H_1	Failure of 1 of 3 HPRS trains with 1 of 2 HTXs
H_1	Failure of 1 of 2 HPRS trains
H_2	Failure of 1 of 2 LPRS trains with 1 of 2 HTXs
H_2	Failure of 1 of 2 LPRS trains
Y_1	Failure of 1 of 4 CCUs
C_1	Failure of 1 of 2 CSIS trains
F	Failure of 1 of 2 CSRS trains with 1 of 2 HTXs
F'	Failure of 1 of 2 CSRS trains

T_3MLQH_1, however, the success events must be considered in the quantification procedure. When every possible core melt accident sequence is traced, there are 64 distinct core melt accident sequences. These accident sequences are given in Table B.6. This table gives the analysis numbers 1 through 64, the core melt accident sequence numbers from the event trees (Reference number), the multiplier events, the Boolean events, the appropriate valve block (VB) to be used in the quantification, and the core melt "failure" accident sequence.

The multiplier events are those events representing systems or functions that are essentially independent of the other events and were not modeled. These events were quantified in Section 3.1. The Boolean events are those events for which fault trees were constructed and must be analyzed using Boolean reduction techniques, namely, the SETS computer program.

An example of a core melt accident sequence reference number is in analysis sequence number 35 where the 45 represents a T_3 initiating event which becomes a transient induced LOCA (that is the 4) and then follows a LOCA branch to the LOCA core melt outcome labeled 5.

The non-Boolean events are grouped in the "multiplier," and the Boolean events are shown spread out so the computer analyst can distinguish patterns to aid in constituting the most efficient SETS runs. Table B.7 delineates the Boolean sequences to be run by SETS and numbered SL #1 through 61 including each unique sequence and applicable value block. As a point of reference, the SETS code is used in this program but will not be discussed here since it is well documented elsewhere.

When the fault trees for all the systems are initially brought together, a value block is used that does not distinguish between the initiating events. The value block is a listing of every basic event by its SETS code name and the assigned probability. When the accident sequences are run, minor adjustments are made to make the value block to fit the initiating event. These adjustments are given in Table B.8.

5.2 Accident Sequence Quantification

It was assumed in the event trees shown in Figure B.7 that the Bleed and Feed mode and the Secondary Blowdown mode were applicable at St. Lucie since operating procedures now exist at the plant. The first subsection describes the quantification of the complete event trees as shown in Figure B.7. The next two subsections address the quantification of the event trees with Bleed and Feed or Secondary Blowdown not applicable. This will show the apparent value of these two modes of operation. The fourth subsection addresses the long term loss of AC power sequence, while the last subsection quantifies the core melt probability if the PORV block valves are normally open.

0279

Table B.6 St. Lucie Core Melt Accident Sequences

Anal. No.	Ref. No.	Multiplier	Boolean Events							VB	Core Melt Failure Sequence
			L	E	D_1	P_1	D	H	H		
1	1	S_2M	$\bar{L}$		$\bar{D}_1$			H'_1	H'_2	1	$S_2MH'_{12}$
2	2	$S_2M\bar{X}$	$\bar{L}$		D_1		$\bar{D}_2$		H'_2	1	$S_2MD_1H'_2$
3	3	$S_2M\bar{X}$	$\bar{L}$		D_1		D_2			1	S_2MD_{12}
4	4	S_2MX	$\bar{L}$		D_1					1	S_2MXD_1
5	5	S_2M	L		$\bar{D}_1$	$\bar{P}_1$		H_1		1	S_2MLH_1
6	6	S_2M	L		$\bar{D}_1$	P_1				1	S_2MLP_1
7	7	S_2M	L		D_1					1	S_2MLD_1
8	2	$T_1M\bar{P}\bar{Q}$	L	$\bar{E}$				H_1		2	T_1MLH_1
9	3	$T_1M\bar{P}\bar{Q}$	L	E						2	T_1MLE
10	5	T_1MP	L							2	T_1MLP
11	2	$T_2M\bar{P}\bar{Q}$	L	$\bar{E}$				H_1		1	T_2MLH_1
12	3	$T_2M\bar{P}\bar{Q}$	L	E						1	T_2MLE
13	5	T_2MP	L							1	T_2MLP
14	2	$T_3M\bar{P}\bar{Q}$	L	$\bar{E}$				H_1		1	T_3MLH_1
15	3	$T_3M\bar{P}\bar{Q}$	L	E						1	T_3MLE
16	5	T_3MP	L							1	T_3MLP
17	11	T_1MQ	$\bar{L}$		$\bar{D}_1$			H'_1	H'_2	2	$T_1MQH'_{12}$
18	12	$T_1MQ\bar{X}$	$\bar{L}$		D_1		$\bar{D}_2$		H'_2	2	$T_1MQD_1H'_2$

0280

Table B.6 St. Lucie Core Melt Accident Sequences (cont.)

Anal. No.	Ref. No.	Multiplier	Boolean Events							VB	Core Melt Failure Sequence
			L	E	D_1	P_1	D_2	H_1	H_2		
19	13	$T_1MQ\bar{X}$	$\bar{L}$		D_1		D_2			2	T_1MQD_{12}
20	14	T_1MQX	$\bar{L}$		D_1					2	T_2MQD_1X
21	45	$T_1M\bar{P}Q$	L		$\bar{D}_1$	$\bar{P}_1$		H_1		2	T_1MQLH_1
22	46	$T_1M\bar{P}Q$	L		$\bar{D}_1$	P_1				2	T_1MQLP_1
23	47	$T_1M\bar{P}Q$	L		D_1					2	T_1MQLD_1
24	11	T_2MQ	$\bar{L}$		$\bar{D}_1$			H'_1	H'_2	1	$T_2MQH'_{12}$
25	12	$T_2MQ\bar{X}$	$\bar{L}$		D_1		$\bar{D}_2$		H'_2	1	$T_2MQD_1H'_2$
26	13	$T_2MQ\bar{X}$	$\bar{L}$		D_1		D_2			1	T_2MQD_{12}
27	14	T_2MQX	$\bar{L}$		D_1					1	T_2MQD_1X
28	45	$T_2M\bar{P}Q$	L		$\bar{D}_1$	$\bar{P}_1$		H_1		1	T_2MQLH_1
29	46	$T_2M\bar{P}Q$	L		$\bar{D}_1$	P_1				1	T_2MQLP_1
30	47	$T_2M\bar{P}Q$	L		D_1					1	T_2MQLD_1
31	11	T_3MQ	$\bar{L}$		$\bar{D}_1$			H'_1	H'_2	1	$T_3MQH'_{12}$
32	12	$T_3MQ\bar{X}$	$\bar{L}$		D_1		$\bar{D}_2$		H'_2	1	$T_3MQD_1H'_2$
33	13	$T_3MQ\bar{X}$	$\bar{L}$		D_1		D_2			1	T_3MQD_{12}
34	14	T_3MQX	$\bar{L}$		D_1					1	T_3MQD_1X
35	45	$T_3M\bar{P}Q$	L		$\bar{D}_1$	$\bar{P}_1$		H_1		1	T_3MQLH_1
36	46	$T_3M\bar{P}Q$	L		$\bar{D}_1$	P_1				1	T_3MQLP_1

Table B.6 St. Lucie Core Melt Accident Sequences (cont.)

Anal. No.	Ref. No.	Multiplier	Boolean Events							VB	Core Melt Failure Sequence
			L	E	D_1	P_1	D_2	H_1	H_2		
37	47	$T_3M\bar{P}Q$	L		D_1					1	T_3MQLD_1
38	61	T_3Q	$\bar{L}$		$\bar{D}_1$			H_1'	H_2'	1	T_3QH_{12}'
39	62	$T_3Q\bar{X}$	$\bar{L}$		D_1		$\bar{D}_2$		H_2'	1	$T_3QD_1H_2'$
40	63	$T_3Q\bar{X}$	$\bar{L}$		D_1		D_2			1	T_3QD_{12}
41	64	T_3QX	$\bar{L}$		D_1					1	T_3QD_1X
42	65	T_3Q	L		$\bar{D}_1$	$\bar{P}_1$		H_1		1	T_3QLH_1
43	66	T_3Q	L		$\bar{D}_1$	P_1				1	T_3QLP_1
44	67	T_3Q	L		D_1					1	T_3QLD_1
45	11	T_4MQ	$\bar{L}$		$\bar{D}_1$			H_1'	H_2'	3	T_4MQH_{12}'
46	12	$T_4MQ\bar{X}$	$\bar{L}$		D_1		$\bar{D}_2$		H_2'	3	$T_4MQD_1H_2'$
47	13	$T_4MQ\bar{X}$	$\bar{L}$		D_1		D_2			3	T_4MQD_{12}
48	14	T_4MQX	$\bar{L}$		D_1					3	T_4MQD_1X
49	45	$T_4M\bar{P}Q$	L		$\bar{D}_1$	$\bar{P}_1$		H_1		3	T_4MQLH_1
50	46	$T_4M\bar{P}Q$	L		$\bar{D}_1$	P_1				3	T_4MQLP_1
51	47	$T_4M\bar{P}Q$	L		D_1					3	T_4MQLD_1
52	2	$T_4M\bar{P}\bar{Q}$	L	$\bar{E}$				H_1		3	T_4MLH_1

0282

Table B.6 St. Lucie Core Melt Accident Sequences (cont.)

Anal. No.	Ref. No.	Multiplier	Boolean Events							VB	Core Melt Failure Sequence
			L	E	D_1	P_1	D_2	H_1	H_2		
53	3	$T_4M\bar{P}\bar{Q}$	L	E						3	T_4MLE
54	5	T_4MP	L							3	T_4MLP
55	11	T_5MQ	$\bar{L}$		$\bar{D}_1$			H'_1	H'_2	4	$T_5MQH'_{12}$
56	12	$T_5MQ\bar{X}$	$\bar{L}$		D_1		$\bar{D}_2$		H'_1	4	$T_5MQD_1H'_2$
57	13	$T_5MQ\bar{X}$	$\bar{L}$		D_1		D_2			4	T_5MQD_{12}
58	14	T_5MQX	$\bar{L}$		D_1					4	T_5MQD_1X
59	45	$T_5M\bar{P}Q$	L		$\bar{D}_1$	$\bar{P}_1$		H_1		4	T_5MQLH_1
60	46	$T_5M\bar{P}Q$	L		$\bar{D}_1$	P_1				4	T_5MQLP_1
61	47	$T_5M\bar{P}Q$	L		D_1					4	T_5MQLD_1
62	2	$T_5M\bar{P}\bar{Q}$	L	$\bar{E}$				H_1		4	T_5MLH_1
63	3	$T_5M\bar{P}\bar{Q}$	L	E						4	T_5MLE
64	5	T_5MP	L							4	T_5MLP

0283

Table B.7 St. Lucie Boolean Accident Sequences

SL No.	L	E	D_1	P_1	D_2	H_1	H_2	VB
1	$\bar{L}$		$\bar{D}_1$			H_1'	H_2'	1
2	$\bar{L}$		D_1		$\bar{D}_2$		H_2'	1
3	$\bar{L}$		D_1		D_2			1
4	$\bar{L}$		D_1					1
5	L		$\bar{D}_1$	$\bar{P}_1$		H_1		1
6	L		$\bar{D}_1$	P_1				1
7	L		D_1					1
8	L							1
9	$\bar{L}$		$\bar{D}_1$			H_1'	H_2'	2
10	$\bar{L}$		D_1		$\bar{D}_2$		H_2'	2
11	$\bar{L}$		D_1		D_2			2
12	$\bar{L}$		D_1					2
13	L		$\bar{D}_1$	$\bar{P}_1$		H_1		2
14	L		$\bar{D}_1$	P_1				2
15	L		D_1					2
16	L	$\bar{E}$				H_1		2
17	L	E						2
18	L							2
19	L	$\bar{E}$				H_1		1
20	L	E						1
21	L							1
22	$\bar{L}$		$\bar{D}_1$			H_1'	H_2'	3
23	$\bar{L}$		D_1		$\bar{D}_2$		H_2'	3
24	$\bar{L}$		D_1		D_2			3
25	$\bar{L}$		D_1					3

0284

Table B.7 St. Lucie Boolean Accident Sequences (cont.)

SL No.	L	E	D_1	P_1	D_2	H_1	H_2	VB
26	L		$\bar{D}_1$	$\bar{P}_1$		H_1		3
27	L		$\bar{D}_1$	P_1				3
28	L		D_1					3
29	L	$\bar{E}$				H_1		3
30	L	E						3
31	L							3
32	$\bar{L}$		$\bar{D}_1$			H_1'	H_2'	4
33	$\bar{L}$		D_1		$\bar{D}_2$		H_2'	4
34	$\bar{L}$		D_1		D_2			4
35	$\bar{L}$		D_1					4
36	L		$\bar{D}_1$	$\bar{P}_1$		H_1		4
37	L		$\bar{D}_1$	P_1				4
38	L		D_1					4
39	L	$\bar{E}$				H_1		4
40	L	E						4
41	L							4
42	L		$\bar{D}_1$	$\bar{P}_1$		H_1		1a
43	L		$\bar{D}_1$	P_1				1a
44	L		$\bar{D}_1$	$\bar{P}_1$		H_1		2a
45	L		$\bar{D}_1$	P_1				2a
46	L	$\bar{E}$				H_1		2a
47	L	E						2a
48	L							2a
49	L	$\bar{E}$				H_1		1a

Table B.7 St. Lucie Boolean Accident Sequences (cont.)

SL No.	L	E	D_1	P_1	D_2	H_1	H_2	VB
50	L	E						1a
51	L							1a
52	L		$\bar{D}_1$	$\bar{P}_1$		H_1		3a
53	L		$\bar{D}_1$	P_1				3a
54	L	$\bar{E}$				H_1		3a
55	L	E						3a
56	L							3a
57	L		$\bar{D}_1$	$\bar{P}_1$		H_1		4a
58	L		$\bar{D}_1$	P_1				4a
59	L	$\bar{E}$				H_1		4a
60	L	E						4a
61	L							4a

Table B.8 Value Block Changes for Accident Sequence Runs

Change Value Block/Input as Follows:

	VB-1	VB-2	VB-3	VB-4
LOSP-OSP	1.0E-3	OMEGA	1.0E-3	1.0E-3
BLKVLV-GATE	OMEGA	OMEGA	OMEGA	OMEGA
BUS-416-1B3-LF	3.0E-5	3.0E-5	OMEGA	3.0E-5
BUS-125-1B-LF	6.0E-5	6.0E-5	6.0E-5	OMEGA
CCW1A-PMS-LF	5.7E-3	8.7E-3	5.7E-3	5.7E-3
CCW1B-PMS-LF	5.7E-3	8.7E-3	5.7E-3	5.7E-3
ICW1A-PMS-LF	5.7E-3	8.7E-3	5.7E-3	5.7E-3
ICW1B-PMS-LF	5.7E-3	8.7E-3	5.7E-3	5.7E-3
BLKVLV-CM	4.0E-4	4.0E-4	4.0E-4	4.0E-4
BLKVLV-1403-LF	8.0E-3	8.0E-3	8.0E-3	8.0E-3
BLKVLV-1405-LF	8.0E-3	8.0E-3	8.0E-3	8.0E-3

	VB-1a	VB-2a	VB-3a	VB-4a
LOSP-OSP	1.0E-3	OMEGA	1.0E-3	1.0E-3
BLKVLV-GATE	0.0	0.0	0.0	0.0
BUS-416-1B3-LF	3.0E-5	3.0E-5	OMEGA	3.0E-5
BUS-125-1B-LF	6.0E-5	6.0E-5	6.0E-5	OMEGA
CCW1A-PMS-LF	5.7E-3	8.7E-3	5.7E-3	5.7E-3
CCW1B-PMS-LF	5.7E-3	8.7E-3	5.7E-3	5.7E-3
ICW1A-PMS-LF	5.7E-3	8.7E-3	5.7E-3	5.7E-3
ICW1B-PMS-LF	5.7E-3	8.7E-3	5.7E-3	5.7E-3
BLKVLV-CM	0.0	0.0	0.0	0.0
BLKVLV-1403-LF	3.6E-5	3.6E-5	3.6E-5	3.6E-5
BLKVLV-1405-LF	3.6E-5	3.6E-5	3.6E-5	3.6E-5

5.2.1 Quantification Including Bleed and Feed and Secondary Blowdown Capability

The results of the SETS runs are given in Table B.9 using a 1E-7 truncation and before recovery was applied. The core melt probabilities are computed by the product of the multiplier frequency and the Boolean sequence probability. Note that several Boolean sequences, designated by the SL#, are used with more than one multiplier. All of the accident sequences with probabilities greater than 1E-6 (marked with *) were carried forward into the recovery analysis. This amounted to 12 sequences which was 97.5% of the core melt probability. The 12 sequences are given in Table B.10.

The recovery analysis methods are given in Section 4.0. In addition to applying the recovery factors, any cut sets which indicated multiple test or maintenance outages of redundant safety related trains were eliminated since they would not be permitted by technical specifications (due to the single failure criterion). After these "recovery" factors were applied to the top 12 accident sequences, the core melt probability was reduced from 1.76E-4 to 1.21E-5. The core melt probabilities with recovery are given in Table B.11 and summarized in Table B.12. The top 12 accident sequences include 98% of the core melt probability after recovery factors have been applied. These accident sequences are the basis for the remaining analysis. Remember that these sequences still give credit for both bleed and feed and secondary blowdown.

5.2.2 Quantification Given Secondary Blowdown Not Applicable

The purpose of this section is to determine the effect of the secondary blowdown mode on core melt probability. To do this, first consider the way the accident sequences would appear without secondary blowdown and what sequences in the original quantification would be replaced by the new accident sequences. This is summarized in Table B.13. In each of the seven cases the new accident sequence replaces the three choices leading to core melt in the secondary blowdown branch by going directly to core melt, i.e., $\bar{L}D_1X$, $\bar{L}D_1\bar{X}D_2$, $\bar{L}D_1\bar{X}\bar{D}_2H'_2$ all reduce to LD_1.

The quantification for each of the accident sequences without secondary blowdown and before technical specification violations were eliminated or credit for recovery actions was included is given in Table B.14 along with the sum of the three sequences it would replace. The net difference is 1.5E-5 at this stage. This analysis was then carried further into the recovery analysis as presented in Table B.15. After recovery, the difference between the sum of the sequences without secondary blowdown and those with is about 2.8E-6.

Table B.9 St. Lucie Base Case Accident Sequence Frequencies Before Recovery (10^{-7} Truncation)

Anal. No.	Multiplier	Boolean Events	SL #	Multiplier Freq.	Boolean Sequence Prob.	Core Melt Prob.		Core Melt Failure Sequence
1	S_2M	$\bar{L}\bar{D}_1H'_1H'_2$	1	2.0E-2	9.7E-4	1.9E-5	*	$S_2MH'_{12}$
2	$S_2M\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	2	2.0E-2	3.2E-6	6.4E-8		$S_2MD_1H'_2$
3	$S_2M\bar{X}$	$\bar{L}D_1D_2$	3	2.0E-2	7.0E-4	1.4E-5	*	S_2MD_{12}
4	S_2MX	$\bar{L}D_1$	4	3.0E-4	1.1E-3	3.3E-7		S_2MXD_1
5	S_2M	$L\bar{D}_1\bar{P}_1H_1$	5	2.0E-2	Φ	Φ		S_2MLH_1
6	S_2M	$L\bar{D}_1P_1$	6	2.0E-2	8.9E-6	1.8E-7		S_2MLP_1
7	S_2M	LD_1	7	2.0E-2	4.4E-7	8.8E-9		S_2MLD_1
8	$T_1M\bar{P}\bar{Q}$	$L\bar{E}H_1$	16	8.3E-2	Φ	Φ		T_1MLH_1
9	$T_1M\bar{P}\bar{Q}$	LE	17	8.3E-2	1.0E-3	8.3E-5	*	T_1MLE
10	T_1MP	L	18	1.1E-7	1.5E-3	1.7E-10		T_1MLP
11	$T_2M\bar{P}\bar{Q}$	LEH_1	19	9.7E-1	Φ	Φ		T_2MLH_1
12	$T_2M\bar{P}\bar{Q}$	LE	20	9.7E-1	9.4E-6	9.1E-6	*	T_2MLE
13	T_2MP	L	21	1.3E-6	5.4E-4	7.0E-10		T_2MLP
14	$T_3M\bar{P}\bar{Q}$	$L\bar{E}H_1$	19	6.9E-2	Φ	Φ		T_3MLH_1
15	$T_3M\bar{P}\bar{Q}$	LE	20	6.9E-2	9.4E-6	6.5E-7		T_3MLE
16	T_3MP	L	21	9.2E-8	5.4E-4	5.0E-11		T_3MLP
17	T_1MQ	$\bar{L}\bar{D}_1H'_1H'_2$	9	1.8E-4	2.4E-3	4.3E-7		$T_1MQH'_{12}$
18	$T_1MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	10	1.7E-4	1.2E-5	2.0E-9		$T_1MQD_1H'_2$

0289

Table B.9 St. Lucie Base Case Accident Sequence Frequencies Before Recovery (10^{-7} Truncation) (cont)

Anal. No.	Multiplier	Boolean Events	SL #	Multiplier Freq.	Boolean Sequence Prob.	Core Melt Prob.		Core Melt Failure Sequence
19	$T_1MQ\bar{X}$	$\bar{L}D_1D_2$	11	1.7E-4	6.9E-3	1.2E-6	*	T_1MQD_{12}
20	T_1MQX	$\bar{L}D_1$	12	6.9E-6	8.0E-3	5.5E-8		T_1MQD_1X
21	$T_1M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	13	2.6E-3	Φ	Φ		T_1MQLH_1
22	$T_1M\bar{P}Q$	$L\bar{D}_1P_1$	14	2.6E-3	2.5E-4	6.5E-7		T_1MQLP_1
23	$T_1M\bar{P}Q$	LD_1	15	2.6E-3	7.7E-4	2.0E-6	*	T_1MQLD_1
24	T_2MQ	$\bar{L}\bar{D}_1H'_1H'_2$	1	2.1E-3	9.7E-4	2.0E-6	*	$T_2MQH'_{12}$
25	$T_2MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	2	2.1E-3	3.2E-6	6.7E-9		$T_2MQD_1H'_2$
26	$T_2MQ\bar{X}$	$\bar{L}D_1D_2$	3	2.1E-3	7.0E-4	1.5E-6	*	T_2MQD_{12}
27	T_2MQX	$\bar{L}D_1$	4	3.2E-5	1.1E-3	3.5E-8		T_2MQD_1X
28	$T_2M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	5	3.0E-2	Φ	Φ		T_2MQLH_1
29	$T_2M\bar{P}Q$	$L\bar{D}_1P_1$	6	3.0E-2	8.9E-6	2.7E-7		T_2MQLP_1
30	$T_2M\bar{P}Q$	LD_1	7	3.0E-2	4.4E-7	1.3E-8		T_2MQLD_1
31	T_3MQ	$\bar{L}\bar{D}_1H'_1H'_2$	1	1.5E-4	9.7E-4	1.5E-7		$T_3MQH'_{12}$
32	$T_3MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	2	1.5E-4	3.2E-6	4.8E-10		$T_3MQD_1H'_2$
33	$T_3MQ\bar{X}$	$\bar{L}D_1D_2$	3	1.5E-4	7.0E-4	1.1E-7		T_3MQD_{12}
34	T_3MQX	$\bar{L}D_1$	4	2.2E-6	1.1E-3	2.4E-9		T_3MQD_1X
35	$T_3M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	5	2.1E-3	Φ	Φ		T_3MQLH_1
36	$T_3M\bar{P}Q$	$L\bar{D}_1P_1$	6	2.1E-3	8.9E-6	1.9E-8		T_3MQLP_1

Table B.9 St. Lucie Base Case Accident Sequence Frequencies Before Recovery (10^{-7} Truncation) (cont)

Anal. No.	Multi-plier	Boolean Events	SL #	Multi-plier Freq.	Boolean Sequence Prob.	Core Melt Prob.		Core Melt Failure Sequence
37	$T_3M\bar{P}Q$	LD_1	7	2.1E-3	4.4E-7	9.2E-10		T_3MQLD_1
38	T_3Q	$\bar{L}\bar{D}_1H'_1H'_2$	1	1.5E-2	9.7E-4	1.5E-5	*	$T_3QH'_{12}$
39	$T_3Q\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	2	1.5E-2	3.2E-6	4.8E-8		$T_3QD_1H'_2$
40	$T_3Q\bar{X}$	$\bar{L}D_1D_2$	3	1.5E-2	7.0E-4	1.1E-5	*	T_3QD_{12}
41	T_3QX	$\bar{L}D_1$	4	2.2E-4	1.1E-3	2.4E-7		T_3QD_1X
42	T_3Q	$L\bar{D}_1\bar{P}_1H_1$	5	1.5E-2	Φ	Φ		T_3QLH_1
43	T_3Q	$L\bar{D}_1P_1$	6	1.5E-2	8.9E-6	1.3E-7		T_3QLP_1
44	T_3Q	LD_1	7	1.5E-2	4.4E-7	6.6E-9		T_3QLD_1
45	T_4MQ	$\bar{L}\bar{D}_1H'_1H'_2$	22	1.3E-5	1.4E-2	1.8E-7		$T_4MQH'_{12}$
46	$T_4MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	23	1.2E-5	1.9E-4	2.3E-9		$T_4MQD_1H'_2$
47	$T_4MQ\bar{X}$	$\bar{L}D_1D_2$	24	1.2E-5	2.5E-2	3.0E-7		T_4MQD_{12}
48	T_4MQX	$\bar{L}D_1$	25	1.9E-7	3.8E-2	7.2E-9		T_4MQD_1X
49	$T_4M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	26	1.8E-4	Φ	Φ		T_4MQLH_1
50	$T_4M\bar{P}Q$	$L\bar{D}_1P_1$	27	1.8E-4	2.0E-3	3.6E-7		T_4MQLP_1
51	$T_4M\bar{P}Q$	LD	28	1.8E-4	6.0E-5	1.1E-8		T_4MQLD_1
52	$T_4M\bar{P}\bar{Q}$	$L\bar{E}H_1$	29	5.8E-3	Φ	Φ		T_4MLH_1
53	$T_4M\bar{P}\bar{Q}$	LE	30	5.8E-3	2.0E-3	1.2E-5	*	T_4MLE
54	T_4MP	L	31	7.8E-9	2.0E-3	1.6E-11		T_4MLP

Table B.9 St. Lucie Base Case Accident Sequence Frequencies Before Recovery (10^{-7} Truncation) (cont)

Anal. No.	Multiplier	Boolean Events	SL #	Multiplier Freq.	Boolean Sequence Prob.	Core Melt Prob.	Core Melt Failure Sequence
55	T_5MQ	$\bar{L}\bar{D}_1H'_1H'_2$	32	6.3E-6	1.0E-3	6.3E-9	$T_5MQH'_{12}$
56	$T_5MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	33	6.2E-6	3.2E-6	2.0E-11	$T_5MQD_1H'_2$
57	$T_5MQ\bar{X}$	$\bar{L}D_1D_2$	34	6.2E-6	7.2E-4	4.5E-9	T_5MQD_{12}
58	T_5MQX	$\bar{L}D_1$	35	9.5E-8	1.1E-3	1.0E-10	T_5MQD_1X
59	$T_5M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	36	9.0E-5	Φ	Φ	T_5MQLH_1
60	$T_5M\bar{P}Q$	$L\bar{D}_1P_1$	37	9.0E-5	2.1E-3	1.9E-7	T_5MQLP_1
61	$T_5M\bar{P}Q$	LD_1	38	9.0E-5	4.9E-5	4.4E-9	T_5MQLD_1
62	$T_5M\bar{P}\bar{Q}$	$L\bar{E}H_1$	39	2.9E-3	Φ	Φ	T_5MLH_1
63	$T_5M\bar{P}\bar{Q}$	LE	40	2.9E-3	2.2E-3	6.4E-6	* T_5MLE
64	T_5MP	L	41	3.9E-9	2.2E-3	8.6E-12	T_5MLP
			All Sequences			1.8067E-4	
			Top 12 Sequences (*) only			1.762E-4	
			Next 52 Sequences			4.469E-6	

0292

Table B.10 St. Lucie Base Case Results Before Recovery Using 1E-7 SETS Truncation - All Accident Sequences Greater Than 1E-6

Analysis Number	Core Melt Failure Sequence	Core Melt Probability	SL Boolean Event Sequence Number
9	T_1MLE	8.3E-5	17
1	$S_2MH_1'H_2'$	1.9E-5	1
38	$T_3QH_1'H_2'$	1.5E-5	1
3	$S_2MD_1D_2$	1.4E-5	3
53	T_4MLE	1.2E-5	30
40	$T_3QD_1D_2$	1.1E-5	3
12	T_2MLE	9.1E-6	20
63	T_5MLE	6.4E-6	40
23	T_1MQLD_1	2.0E-6	15
24	$T_2MQH_1'H_2'$	2.0E-6	1
26	$T_2MQD_1D_2$	1.5E-6	3
19	$T_1MQD_1D_2$	1.2E-6	11
		Sum = 1.762E-4	

0293

Table B.11 St. Lucie Base Case Accident Sequence Frequencies After Recovery (10^{-7} Truncation)

Anal. No.	Multi-plier	Boolean Events	SL #	Multi-plier Freq.	Boolean Sequence Prob.	Core Melt Prob.	Core Melt Failure Sequence
1	S_2M	$\bar{L}\bar{D}_1H'_1H'_2$	1	2.0E-2	5.5E-5	1.1E-6	* $S_2MH'_{12}$
2	$S_2M\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	2				$S_2MD_1H'_2$
3	$S_2M\bar{X}$	$\bar{L}D_1D_2$	3	2.0E-2	9.5E-5	1.9E-6	* S_2MD_{12}
4	S_2MX	$\bar{L}D_1$	4				S_2MXD_1
5	S_2M	$L\bar{D}_1\bar{P}_1H_1$	5			Φ	S_2MLH_1
6	S_2M	$L\bar{D}_1P_1$	6			1.8E-8	S_2MLP_1
7	S_2M	LD_1	7			7.9E-9	S_2MLD_1
8	$T_1M\bar{P}\bar{Q}$	$L\bar{E}H_1$	16				T_1MLH_1
9	$T_1M\bar{P}\bar{Q}$	LE	17	8.3E-2	9.5E-5	7.9E-6	* T_1MLE
10	T_1MP	L	18			3.4E-11	T_1MLP
11	$T_2M\bar{P}\bar{Q}$	$L\bar{E}H_1$	19			Φ	T_2MLH_1
12	$T_2M\bar{P}\bar{Q}$	LE	20	9.7E-1	2.0E-7	1.9E-7	T_2MLE
13	T_2MP	L	21			2.1E-11	T_2MLP
14	$T_3M\bar{P}\bar{Q}$	$L\bar{E}H_1$	19				T_3MLH_1
15	$T_3M\bar{P}\bar{Q}$	LE	20				T_3MLE
16	T_3MP	L	21				T_3MLP
17	T_1MQ	$\bar{L}\bar{D}_1H'_1H'_2$	9			4.3E-8	$T_1MQH'_{12}$
18	$T_1MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	10			2.0E-10	$T_1MQD_1H'_2$

0294

Table B.11 St. Lucie Base Case Accident Sequence Frequencies After Recovery (10^{-7} Truncation) (cont)

Anal. No.	Multi- plier	Boolean Events	SL #	Multi- plier Freq.	Boolean Sequence Prob.	Core Melt Prob.	Core Melt Failure Sequence
19	$T_1MQ\bar{X}$	$\bar{L}D_1D_2$	11	1.7E-4	1.1E-3	1.9E-7	T_1MQD_{12}
20	T_1MQX	$\bar{L}D_1$	12			1.7E-8	T_1MQD_1X
21	$T_1M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	13			Φ	T_1MQLH_1
22	$T_1M\bar{P}Q$	$L\bar{D}_1P_1$	14			8.0E-8	T_1MQLP_1
23	$T_1M\bar{P}Q$	LD_1	15	2.6E-3	1.1E-4	2.9E-7	T_1MQLD_1
24	T_2MQ	$\bar{L}\bar{D}_1H'_1H'_2$	1	2.1E-3	5.5E-5	1.2E-7	$T_2MQH'_{12}$
25	$T_2MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	2				$T_2MQD_1H'_2$
26	$T_2MQ\bar{X}$	$\bar{L}D_1D_2$	3	2.1E-3	8.3E-5	1.7E-7	T_2MQD_{12}
27	T_2MQX	$\bar{L}D_1$	4				T_2MQD_1X
28	$T_2M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	5			Φ	T_2MQLH_1
29	$T_2M\bar{P}Q$	$L\bar{D}_1P_1$	6			2.7E-8	T_2MQLP_1
30	$T_2M\bar{P}Q$	LD_1	7			1.3E-9	T_2MQLD_1
31	T_3MQ	$\bar{L}\bar{D}_1H'_1H'_2$	1				$T_3MQH'_{12}$
32	$T_3MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	2				$T_3MQD_1H'_2$
33	$T_3MQ\bar{X}$	$\bar{L}D_1D_2$	3				T_3MQD_{12}
34	T_3MQX	$\bar{L}D_1$	4				T_3MQD_1X
35	$T_3M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	5				T_3MQLH_1
36	$T_3M\bar{P}Q$	$L\bar{D}_1P_1$	6				T_3MQLP_1

Table B.11 St. Lucie Base Case Accident Sequence Frequencies After Recovery (10^{-7} Truncation) (cont)

Anal. No.	Multi- plier	Boolean Events	SL #	Multi- plier Freq.	Boolean Sequence Prob.	Core Melt Prob.	Core Melt Failure Sequence
37	$T_3M\bar{P}Q$	LD_1	7				T_3MQLD_1
38	T_3Q	$\bar{L}\bar{D}_1H'_1H'_2$	1	1.5E-2	5.5E-5	8.3E-7	$T_3QH'_{12}$
39	$T_3Q\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	2				$T_3QD_1H'_2$
40	$T_3Q\bar{X}$	$\bar{L}D_1D_2$	3	1.5E-2	8.3E-5	1.2E-6	* T_3QD_{12}
41	T_3QX	$\bar{L}D_1$	4				T_3QD_1X
42	T_3Q	$L\bar{D}_1\bar{P}_1H_1$	5			Φ	T_3QLH_1
43	T_3Q	$L\bar{D}_1P_1$	6			3.9E-9	T_3QLP_1
44	T_3Q	LD_1	7			6.6E-10	T_3QLD_1
45	T_4MQ	$\bar{L}\bar{D}_1H'_1H'_2$	22				$T_4MQH'_{12}$
46	$T_4MQ\bar{X}$	$\bar{L}D_1\bar{D}_2H'_2$	23				$T_4MQD_1H'_2$
47	$T_4MQ\bar{X}$	$\bar{L}D_1D_2$	24				T_4MQD_{12}
48	T_4MQX	$\bar{L}D_1$	25				T_4MQD_1X
49	$T_4M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	26			Φ	T_4MQLH_1
50	$T_4M\bar{P}Q$	$L\bar{D}_1P_1$	27			3.6E-8	T_4MQLP_1
51	$T_4M\bar{P}Q$	LD	28			1.1E-9	T_4MQLD_1
52	$T_4M\bar{P}\bar{Q}$	$L\bar{E}H_1$	29			Φ	T_4MLH_1
53	$T_4M\bar{P}\bar{Q}$	LE	30	5.8E-3	3.0E-5	1.7E-7	T_4MLE
54	T_4MP	L	31			4.8E-13	T_4MLP

0296

Table B.11 St. Lucie Base Case Accident Sequence Frequencies After Recovery (10^{-7} Truncation) (cont)

Anal. No.	Multi-plier	Boolean Events	SL #	Multi plier Freq.	Boolean Sequence Prob.	Core Melt Prob.	Core Melt Failure Sequence
55	T_5MQ	$\bar{L}\bar{D}_1H'_1H'_2$	32				$T_5MQH'_{12}$
56	$T_5M\bar{Q}X$	$\bar{L}D_1\bar{D}_2H'_2$	43				$T_5MQD_1H'_2$
57	$T_5MQ\bar{X}$	$\bar{L}D_1D_2$	34				T_5MQD_{12}
58	T_5MQX	$\bar{L}D_1$	35				T_5MQD_1X
59	$T_5M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	36			Φ	T_5MQLH_1
60	$T_5M\bar{P}Q$	$L\bar{D}_1P_1$	37			1.9E-8	T_5MQLP_1
61	$T_5M\bar{P}Q$	LD_1	38			4.4E-10	T_5MQLD_1
62	$T_5M\bar{P}\bar{Q}$	$L\bar{E}H_1$	39			Φ	T_5MLH_1
63	$T_5M\bar{P}\bar{Q}$	LE	40	2.9E-3	1.4E-5	4.1E-8	T_5MLE
64	T_5MP	L	41			2.6E-13	T_5MLP

Sum = 1.436E-5

Top 12 Sequences (*) only 1.41E-5

Next 27 Sequences 2.56E-7

0297

Table B.12 St. Lucie Base Case Results After Recovery - Top 12 Sequences

Analysis Number	Core Melt Failure Sequence	Core Melt Frequency	SL Boolean Event Sequence Number
9	T_1MLE	7.9E-6	17
3	$S_2MD_1D_2$	1.9E-6	3
40	$T_3QD_1D_2$	1.2E-6	3
1	$S_2MH_1'H_2'$	1.1E-6	1
38	$T_3QH_1'H_2'$	8.3E-7	1
23	T_1MQLD_1	2.9E-7	15
19	$T_1MQD_1D_2$	1.9E-7	11
12	T_2MLE	1.9E-7	20
26	$T_2MQD_1D_2$	1.7E-7	3
53	T_4MLE	1.7E-7	30
24	$T_2MQH_1'H_2'$	1.2E-7	1
63	T_5MLE	4.1E-8	40

Sum = 1.41E-5

0298

Table B.13 St. Lucie Core Melt Accident Sequences Without Secondary Blowdown

Anal. No.	Multi-plier	Boolean Events	SL Seq. No.	Replaces Basecase Sequence Analysis Numbers
65	S_2M	$\bar{L}D_1$	4	2, 3, and 4
66	T_1MQ	$\bar{L}D_1$	12	18, 19, and 20
67	T_2MQ	$\bar{L}D_1$	4	25, 26, and 27
68	T_3MQ	$\bar{L}D_1$	4	32, 33, and 34
69	T_3Q	$\bar{L}D_1$	4	39, 40, and 41
70	T_5MQ	$\bar{L}D_1$	25	46, 47, and 48
71	T_5MQ	$\bar{L}D_1$	35	56, 57, and 58

0299

Table B.14 St. Lucie Core Melt Accident Sequences Probabilities Without Secondary Blowdown and Before Recovery

Anal. No.	Multi-plier	Boolean Sequence	SL Seq. No.	Multi. Freq.	Boolean Sequence Freq.	Core Melt Freq.	Sum of Core Melt Frequencies for Sequences Replaced
65	S_2M	$\bar{L}D_1$	4	2.0E-2	1.1E-3	2.2E-5	1.4E-5
66	T_1MQ	$\bar{L}D_1$	12	1.8E-4	8.0E-3	1.4E-6	1.3E-6
67	T_2MQ	$\bar{L}D_1$	4	2.1E-3	1.1E-3	2.3E-6	1.5E-6
68	T_3MQ	$\bar{L}D_1$	4	1.5E-4	1.1E-3	1.7E-7	1.1E-7
69	T_3Q	$\bar{L}D_1$	4	1.5E-2	1.1E-3	1.7E-5	1.1E-5
70	T_4MQ	$\bar{L}D_1$	25	1.3E-5	3.8E-2	4.9E-7	3.1E-7
71	T_5MQ	$\bar{L}D$	35	6.3E-6	1.1E-3	6.9E-9	4.6E-9
					Sum =	4.34E-5	2.82E-5

0300

Table B.15 St. Lucie Core Melt Accident Sequences Probabilities Without Secondary Blowdown and After Recovery

Anal. No.	Multi-plier	Boolean Sequence	SL Seq. No.	Multi. Freq.	Boolean Sequence Freq.	Core Melt Freq.	Sum of Core Melt Frequencies for Sequences Replaced
65	S_2M	$\bar{L}D_1$	4	2.0E-2	1.7E-4	3.4E-6	1.9E-6
66	T_1MQ	$\bar{L}D_1$	12	1.8E-4	1.2E-3	2.2E-7	2.1E-7
67	T_2MQ	$\bar{L}D_1$	4	2.1E-3	1.5E-4	3.2E-7	1.7E-7
68	T_3MQ	$\bar{L}D_1$	4	1.5E-4	1.5E-4	2.3E-8	---
69	T_3Q	$\bar{L}D_1$	4	1.5E-2	1.5E-4	2.3E-6	1.2E-6
70	T_4MQ	$\bar{L}D_1$	25				
71	T_5MQ	$\bar{L}D_1$	35				
					Sum =	6.26E-6	3.48E-6

0301

5.2.3 Quantification Given Bleed and Feed Not Applicable

The purpose of this section is to determine the effect of the bleed and feed mode on core melt probability. This is accomplished by considering the way the accident sequences would appear without bleed and feed, and which of the original sequences would be replaced by the new accident sequences. This is summarized in Table B.16. In each of the first seven cases (i.e., analysis numbers 72 through 78) there are three choices leading to core melt in the branch, i.e., LD_1, $L\bar{D}_1P_1$ and $L\bar{D}_1\bar{P}_1H_1$ replaced by L alone. In the last five cases which are solely transients, that is no induced LOCA, the Boolean sequences $L\bar{P}\bar{Q}E$ and $L\bar{P}\bar{Q}\bar{E}H_1$ are replaced by $L\bar{P}\bar{Q}$. This is essentially bleed without feed and would eventually lead to core uncovery while the pressure remains too high for safety injection but not high enough to result in reactor vessel rupture. This seems rather artificial since it is hard to imagine a plant not attempting to reduce the pressure using the PORVs and injecting which, of course, is bleed and feed! Nevertheless if the plant has insufficient capability perhaps over the long term it could lead to core melt.

The quantification for each of the accident sequences without bleed and feed and before credit for Tech Specs or recovery are included is given in Table B.17 along with the corresponding sum of the sequences it would replace. The net difference is 5.83E-4. The quantification for these same sequences after "recovery" is applied is given in Table B.18. The net difference here is 3.39E-5. Since emergency operating procedures are now in place at St. Lucie for a bleed and feed mode, these replacement sequences (72 through 83) are not the sequences used in the calculation of total plant core melt for the base case.

If the sequences numbered 65 through 83 are substituted for the original sequences which included both secondary blowdown and bleed and feed, we would be left with a sensitivity analysis for calculating the core melt probability for a plant without bleed and feed or secondary blowdown. These sequences are presented in Table B.19 with a total core melt probability after recovery of 5.39E-5.

5.2.4 Long Term Station Blackout Analysis

Following a loss of offsite power with a subsequent failure of the power conversion system, immediate failure of the diesel generators will fail the high pressure injection pumps as well as the ability to recharge the batteries. Under these conditions, decay heat removal would be provided via the DC dependent turbine driven AFW pump until the batteries are depleted or the source of water is emptied.

Table B.16 St. Lucie Core Melt Accident Sequences Without Bleed and Feed

Anal. No.	Multiplier	Boolean Events	SL Seq. No.	Replaces Basecase Sequence Analysis Numbers
72	S_2M	L	8	5, 6, and 7
73	$T_1M\bar{P}Q$	L	18	21, 22, and 23
74	$T_2M\bar{P}Q$	L	21	28, 29, and 30
75	$T_3M\bar{P}Q$	L	21	35, 36, and 37
76	T_3Q	L	21	42, 43, and 44
77	$T_5M\bar{P}Q$	L	31	49, 50, and 51
78	$T_5M\bar{P}Q$	L	41	59, 60, and 61
79	$T_1M\bar{P}\bar{Q}$	L	18	8 and 9
80	$T_2M\bar{P}\bar{Q}$	L	21	11 and 12
81	$T_3M\bar{P}\bar{Q}$	L	21	14 and 15
82	T_4MPQ	L	31	52 and 53
83	$T_5M\bar{P}\bar{Q}$	L	41	62 and 63

Table B.17 St. Lucie Core Melt Accident Sequence Probabilities Without Bleed and Feed and Before Recovery

Anal. No.	Multi- plier	Boolean Sequence	SL Seq. No.	Multi. Freq.	Boolean Sequence Freq.	Core Melt Freq.	Sum of Core Melt Frequencies for Sequences Replaced
72	S_2M	L	8	2.0E-2	5.4E-4	1.1E-5	1.9E-7
73	$T_1M\bar{P}Q$	L	18	2.6E-3	1.0E-3	2.6E-6	2.7E-6
74	$T_2M\bar{P}Q$	L	21	3.0E-2	5.4E-4	1.6E-5	2.8E-7
75	$T_3M\bar{P}Q$	L	21	2.1E-3	5.4E-4	1.1E-6	2.0E-8
76	T_3Q	L	21	1.5E-2	5.4E-4	8.1E-6	1.4E-7
77	$T_4M\bar{P}Q$	L	31	1.8E-4	2.0E-3	3.6E-7	3.7E-7
78	$T_5M\bar{P}Q$	L	41	9.0E-5	2.2E-3	2.0E-7	1.9E-7
79	$T_1M\bar{P}\bar{Q}$	L	18	8.3E-2	1.0E-3	8.3E-5	8.3E-5
80	$T_2M\bar{P}\bar{Q}$	L	21	9.7E-1	5.4E-4	5.2E-4	9.1E-6
81	$T_3M\bar{P}\bar{Q}$	L	21	6.9E-2	5.4E-4	3.7E-5	6.5E-7
82	$T_4M\bar{P}\bar{Q}$	L	31	5.8E-3	2.0E-3	1.2E-5	1.2E-5
83	$T_5M\bar{P}\bar{Q}$	L	41	2.9E-3	2.2E-3	6.4E-6	6.4E-6
					Sum =	6.98E-4	1.15E-4

Table B.18 St. Lucie Core Melt Accicent Sequence Probabilities Without Bleed and Feed and After Recovery

Anal. No.	Multi- plier	Boolean Sequence	SL Seq. No.	Multi. Freq.	Boolean Sequence Freq.	Core Melt Freq.	Sum of Core Melt Frequencies for Sequences Replaced
72	S_2M	L	8	2.0E-2	2.0E-4	4.0E-6	2.6E-8
73	$T_1M\bar{P}Q$	L	18	2.6E-3	2.0E-4	5.2E-7	3.7E-7
74	$T_2M\bar{P}Q$	L	21	3.0E-2	2.2E-5	6.6E-7	2.8E-8
75	$T_3M\bar{P}Q$	L	21	2.1E-3	2.2E-4	4.6E-7	-
76	T_3Q	L	21	1.5E-2	2.2E-5	3.3E-7	4.6E-9
77	$T_4M\bar{P}Q$	L	31	1.8E-4	3.0E-5	5.4E-9	-
78	$T_5M\bar{P}Q$	L	41	9.0E-5	1.4E-5	1.3E-9	-
79	$T_1M\bar{P}\bar{Q}$	L	18	8.3E-2	1.8E-4	1.5E-5	7.9E-6
80	$T_2M\bar{P}\bar{Q}$	L	21	9.7E-1	6.6E-6	6.4E-6	1.9E-7
81	$T_3M\bar{P}\bar{Q}$	L	21	6.9E-2	2.2E-4	1.5E-5	-
82	$T_4M\bar{P}\bar{Q}$	L	31	5.8E-3	3.0E-5	1.7E-7	1.7E-7
83	$T_5M\bar{P}\bar{Q}$	L	41	2.9E-3	1.4E-5	4.1E-8	4.1E-8
					Sum =	4.26E-5	8.73E-6

Table B.19 St. Lucie Core Melt Accicent Sequences Without Secondary Blowdown or Bleed and Feed and After Recovery (1E-7 Truncation)

Anal. No.	Multi-plier	Boolean Events	Multi-plier Freq.	Boolean Sequence Freq.	Core Melt Prob.	Core Melt Sequences Replaced
1	S_2M	$\bar{L}\bar{D}_1H'_1H'_2$	2.0E-2	5.5E-5	1.1E-6	$S_2MH'_{12}$
65	S_2M	$\bar{L}D_1$	2.0E-2	1.7E-4	3.4E-6	S_2MD_1
72	S_2M	L	2.0E-2	2.0E-4	4.0E-6	S_2ML
79	$T_1M\bar{P}\bar{Q}$	L	8.3E-2	1.8E-4	1.5E-5	T_1ML
10	T_1MP	L		ε		T_1MLP
80	$T_2M\bar{P}\bar{Q}$	L	9.7E-1	6.6E-6	6.4E-6	T_2ML
13	T_2MP	L		ε		T_2MLP
81	$T_3M\bar{P}\bar{Q}$	L	6.9E-2	2.2E-4	1.5E-5	T_3ML
16	T_3MP	L		ε		T_3MLP
17	T_1MQ	$\bar{L}\bar{D}_1H'_1H'_2$		ε		$T_1MQH'_{12}$
66	T_1MQ	$\bar{L}D_1$	1.8E-4	1.2E-3	2.2E-7	T_1MQD_1
73	$T_1M\bar{P}Q$	L	2.6E-3	2.0E-4	5.2E-7	T_1MQL
24	T_2MQ	$\bar{L}\bar{D}_1H'_1H'_2$	2.1E-3	5.5E-5	1.2E-7	$T_2MQH'_{12}$
67	T_2MQ	$\bar{L}D_1$	2.1E-3	1.5E-4	3.2E-7	T_2MQD_1
74	$T_2M\bar{P}Q$	L	3.0E-2	2.2E-5	6.6E-7	T_2MQL
31	T_3MQ	$\bar{L}\bar{D}_1H'_1H'_2$		ε		$T_3MQH'_{12}$
68	T_3MQ	$\bar{L}D_1$	1.5E-4	1.5E-4	2.3E-8	T_3MQD_1
75	$T_3M\bar{P}Q$	L	2.1E-3	2.2E-4	4.6E-7	T_3MQL

Table B.19 St. Lucie Core Melt Accicent Sequences Without Secondary Blowdown or Bleed and Feed and After Recovery (1E-7 Truncation) (cont.)

Anal. No.	Multi-plier	Boolean Events	Multi-plier Freq.	Boolean Sequence Freq.	Core Melt Prob.	Core Melt Sequences Replaced
38	T_3Q	$\bar{L}\bar{D}_1H'_1H'_2$	1.5E-2	5.5E-5	8.3E-7	$T_3QH'_{12}$
69	T_3Q	$\bar{L}D_1$	1.5E-2	1.5E-4	2.3E-6	T_3QD_1
76	T_3Q	L	1.5E-2	2.2E-4	3.3E-6	T_3QL
45	T_4MQ	$\bar{L}\bar{D}_1H'_1H'_2$		ε		$T_4MQH'_{12}$
70	T_4MQ	$\bar{L}D_1$		ε		T_4MQD_1
77	$T_4M\bar{P}Q$	L	1.8E-4	3.0E-5	5.4E-9	T_4MQL
82	$T_4M\bar{P}\bar{Q}$	L	5.8E-3	3.0E-5	1.7E-7	T_4ML
54	T_4MP	L		ε		T_4MLP
55	T_5MQ	$\bar{L}\bar{D}_1H'_1H'_2$		ε		$T_5MQH'_{12}$
71	T_5MQ	$\bar{L}D_1$		ε		T_5MQD_1
78	$T_5M\bar{P}Q$	L	9.0E-5	1.4E-5	1.3E-9	T_5MQL
83	$T_5M\bar{P}\bar{Q}$	L	2.9E-3	1.4E-5	4.1E-8	T_5ML
64	T_5MP	L		ε		T_5MLP
				Sum =	5.39E-5	

0307

The turbine driven pump is capable of supplying auxiliary feedwater flow to the steam generators for the total expected range of steam generator pressure (985 psig to 50 psig) by means of a variable speed turbine driver controlled by a variable speed hydraulic governor. The pump relies solely on a DC power supply; the valves associated with the turbine driven pump also are powered from a DC source (see Figure B.8). The turbine driven pump supplies feedwater to both steam generators by means of two separate lines each with its own control valve. The control of auxiliary feedwater flow and steam generator level is normally accomplished by means of control-room operated control valves, each sized to pass the full flow. A local control system is provided to facilitate shutdown if the control room is not accessible.

St. Lucie has two 125V lead-calcium station safety-grade batteries, each of which consists of 60 cells and is rated for 1800 ampere-hour at an 8 hour discharge rate. This rating is sufficient for an 8 hour emergency period without assistance from a battery charger with all non-emergency loads disconnected within one hour.

NUREG-1109 entitled Regulatory Analysis for the Resolution of Unresolved Safety Issue A-44, Station Blackout has concluded that 8 hour batteries are sufficient to ensure DC power for long term cooling.

The storage capacity of the condensate storage tank is 250,000 gallons, a sufficient quantity to meet the requirements for decay heat removal and cooldown of the nuclear steam supply system. After a loss of normal feedwater, approximately 110,000 gallons of water are required to permit cooldown to 325°F following a reactor trip. A crosstie is provided between the Unit 1 auxiliary feedwater pumps and the Unit 2 condensate storage tank to add additional system flexibility.

Analysis indicated that 138,000 gallons are used for 8 hours of hot standby following a reactor trip. This includes 28,000 gallons to raise the water level to the normal operating level from the low level it would be at because of the time lapse in initiation of the auxiliary feedwater system following loss of main feed pumps.

Therefore, it is concluded that St. Lucie has sufficient turbine driven AFW pump flow capacity, electric power, and condensate storage water to assure that station blackout is not a concern.

5.2.5 Quantification Given PORV Block Valves are Normally Open

The purpose of this section is to determine the effect of the normal PORV block valve position on core melt probability. The base case analysis uses the assumption that these block valves

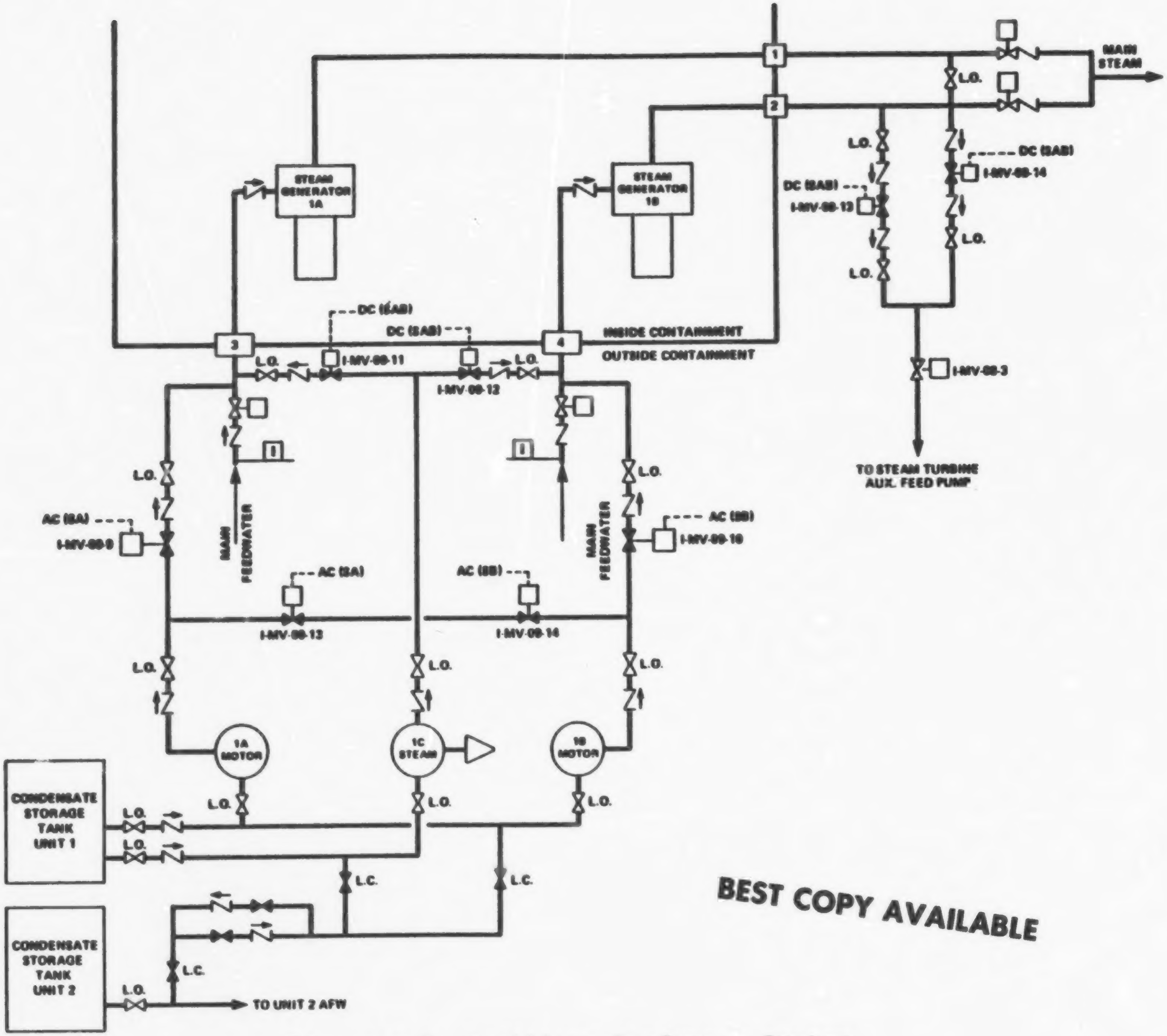
MAIN STEAM
1
2
L.O.
STEAM GENERATOR 1A
STEAM GENERATOR 1B
L.O.
DC (SAB)
I-MV-09-14
DC (SAB)
I-MV-09-13
L.O.
L.O.
I-MV-09-3
TO STEAM TURBINE AUX. FEED PUMP
DC (SAB)
DC (SAB)
3
4
INSIDE CONTAINMENT
OUTSIDE CONTAINMENT
L.O.
I-MV-09-11
I-MV-09-12
L.O.
L.O.
AC (SA)
I-MV-09-9
MAIN FEEDWATER
AC (SA)
I-MV-09-13
AC (SB)
I-MV-09-14
MAIN FEEDWATER
L.O.
AC (SB)
I-MV-09-10
L.O.
L.O.
L.O.
1A MOTOR
1C STEAM
1B MOTOR
L.O.
L.O.
L.O.
CONDENSATE STORAGE TANK UNIT 1
L.O.
L.O.
L.C.
L.C.
CONDENSATE STORAGE TANK UNIT 2
L.C.
L.O.
TO UNIT 2 AFW

Figure B.8 Auxiliary Feedwater System

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0309

will be normally closed to prevent PORV leakage. Although this is more conservative than modeling the block valves as normally open, it is not unrealistic. For most of the plants examined in this study, at least one of the block valves was, indeed, kept closed. This configuration requires electric power and a signal to open the block valves when pressure is to be relieved. If the block valves are assumed to be normally open, neither electric power or a signal are necessary to open them. This reduces the probability of their failure and consequently reduces the core melt contribution. Table B.20 lists the sequences affected by the position of the block valves.

Table B.21 gives the quantification of the sequences with and without blocked PORVs before recovery is applied. The core melt probability for just these affected sequences is reduced from 1.13E-4 to 8.09E-5 by unblocking the PORVs. Table B.22 gives the results for those accident sequences after recovery has been applied to the 4 sequences in Table B.21 which were approximately 1E-6 or greater. The final result is a core melt reduction from 8.30E-6 to 7.24E-6 if the block valves are assumed to be normally open. Thus, the conservative assumption in the base case that the PORV block valves are normally closed (and need electric power to open) increases the core melt probability by approximately 1.1E-6/R-yr.

5.3 Vulnerability Identification

First, the cut sets were examined to identify the specific internal vulnerabilities, and then later the dominant accident sequences were combined with the containment systems event tree and thus extrapolated to public risk measures later in the analysis.

The specific vulnerabilities were identified using Boolean event sequences derived from the SETS program. The top 6 accident sequences result from 4 Boolean event sequences. The raw output for these 4 Boolean event sequences showing the dominant cut sets is given in Table B.23. It can be seen that Boolean sequence 3 had recovery applied in more than one analysis depending upon the initiating event and resultant time available to recover. Each of the top 6 accident sequences were examined for the most significant cut sets. Note that the recovery, RA-VCCF, is applied to several cut sets where the VCC-LF event is not present. This RA opens the steam input MOVs to the AFW system allowing it to function, thus circumventing the battery failure. Table B.24 tabulates these cut sets and their core melt frequencies with the residual core melt frequency for each accident sequence contributed by all the other cut sets in that accident sequence that are not considered further.

Table B.20 St. Lucie Core Melt Accident Sequences with Bleed and Feed with the PORVs Unblocked

Anal. No.	Multiplier	Boolean Events	SL Seq. No.	Replaces Basecase Seq. Analysis No.
84	S_2M	$L\bar{D}_1\bar{P}_1H_1$	42	5
85	S_2M	$L\bar{D}_1P_1$	43	6
86	$T_1M\bar{P}Q$	$LD_1P_1H_1$	44	21
87	$T_1M\bar{P}Q$	$L\bar{D}_1P_1$	45	22
88	$T_2M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	42	28
89	$T_2M\bar{P}Q$	$L\bar{D}_1P_1$	43	29
90	$T_3M\bar{P}Q$	$LD_1P_1H_1$	42	35
91	$T_3M\bar{P}Q$	$L\bar{D}_1P_1$	43	36
92	T_3Q	$L\bar{D}_1\bar{P}_1H_1$	42	42
93	T_3Q	$L\bar{D}_1P_1$	43	43
94	$T_4M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	52	49
95	$T_4M\bar{P}Q$	$L\bar{D}_1P_1$	53	50
96	$T_5M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	57	59
97	$T_5M\bar{P}Q$	$L\bar{D}_1P_1$	58	60
98	$T_1M\bar{P}\bar{Q}$	$L\bar{E}H_1$	46	8
99	$T_1M\bar{P}\bar{Q}$	LE	47	9
100	$T_2M\bar{P}\bar{Q}$	$L\bar{E}H_1$	49	11
101	$T_2M\bar{P}\bar{Q}$	LE	50	12

Table B.20 St. Lucie Core Melt Accident Sequences with Bleed and Feed with the PORVs Unblocked (cont.)

Anal. No.	Multiplier	Boolean Events	SL Seq. No.	Replaces Basecase Seq. Analysis No.
102	$T_3M\bar{P}\bar{Q}$	$L\bar{E}H_1$	49	14
103	$T_3M\bar{P}\bar{Q}$	LE	50	15
104	$T_4M\bar{P}\bar{Q}$	$L\bar{E}H_1$	54	52
105	$T_4M\bar{P}\bar{Q}$	LE	55	53
106	$T_5M\bar{P}\bar{Q}$	$L\bar{E}H_1$	59	62
107	$T_5M\bar{P}\bar{Q}$	LE	60	63
108	T_1MP	L	48	10
109	T_2MP	L	51	13
110	T_3MP	L	51	16
111	T_4MP	L	56	54
112	T_5MP	L	61	64

Table B.21 St. Lucie Core Melt Accident Sequence Probabilities for Bleed and Feed with the PORVs Unblocked Before Recovery

Anal. No.	Multi-plier	Boolean Sequence	SL Seq. No.	Multi. Freq.	Boolean Sequence Freq.	Core Melt Freq.	Core Melt Frequencies for Sequences Replaced
84	S_2M	$L\bar{D}_1\bar{P}_1H_1$	42	2.0E-4	Φ	Φ	Φ
85	S_2M	$L\bar{D}_1P_1$	43	2.0E-2	1.2E-6	2.4E-8	1.8E-7
86	$T_1M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	44	2.6E-3	Φ	Φ	Φ
87	$T_1M\bar{P}Q$	$L\bar{D}_1P_1$	45	2.6E-3	8.2E-5	2.1E-7	6.5E-7
88	$T_2M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	42	3.0E-2	Φ	Φ	Φ
89	$T_2M\bar{P}Q$	$L\bar{D}_1P_1$	43	3.0E-2	1.2E-6	3.6E-8	2.7E-7
90	$T_3M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	42	2.1E-3	Φ	Φ	Φ
91	$T_3M\bar{P}Q$	$L\bar{D}_1P_1$	43	2.1E-3	1.2E-6	2.5E-9	1.9E-8
92	T_3Q	$L\bar{D}_1\bar{D}_1H_1$	42	1.5E-2	Φ	Φ	Φ
93	T_3Q	$L\bar{D}_1P_1$	43	1.5E-2	1.2E-6	1.8E-8	1.3E-7
94	$T_4M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	52	1.8E-4	5.1E-5	9.2E-9	Φ
95	$T_4M\bar{P}Q$	$L\bar{D}_1P_1$	53	1.8E-4	9.6E-5	1.7E-8	3.6E-7
96	$T_5M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	57	9.0E-5	Φ	Φ	Φ
97	$T_5M\bar{P}Q$	$L\bar{D}_1P_1$	58	9.0E-5	2.1E-3	1.9E-7	1.9E-7
98	$T_1M\bar{P}\bar{Q}$	$L\bar{E}H_1$	46	8.3E-2	1.0E-7	8.3E-9	Φ
99	$T_1M\bar{P}\bar{Q}$	LE	47	8.3E-2	8.5E-4	7.1E-5	* 8.3E-5
100	$T_2M\bar{P}\bar{Q}$	$L\bar{E}H_1$	49	9.7E-1	Φ	Φ	Φ

0313

Table B.21 St. Lucie Core Melt Accident Sequence Probabilities for Bleed and Feed with the PORVs Unblocked Before Recovery (cont.)

Anal. No.	Multi-plier	Boolean Sequence	SL Seq. No.	Multi. Freq.	Boolean Sequence Freq.	Core Melt Freq.	Core Melt Frequencies for Sequences Replaced
101	$T_2M\bar{P}\bar{Q}$	LE	50	9.7E-1	1.6E-6	1.6E-6	* 9.1E-6
102	$T_3M\bar{P}\bar{Q}$	$L\bar{E}H_1$	49	6.9E-2	Φ	Φ	Φ
103	$T_3M\bar{P}\bar{Q}$	LF	50	6.9E-2	1.6E-6	1.1E-7	6.5E-7
104	$T_4M\bar{P}\bar{Q}$	$L\bar{E}H_1$	54	5.8E-3	7.8E-5	4.5E-7	Φ
105	$T_4M\bar{P}\bar{Q}$	LE	55	5.8E-3	1.5E-4	8.7E-7	* 1.2E-5
106	$T_5M\bar{P}\bar{Q}$	$L\bar{E}H_1$	59	2.9E-3	Φ	Φ	Φ
107	$T_5M\bar{P}\bar{Q}$	LE	60	2.9E-3	2.2E-3	6.4E-6	* 6.4E-6
108	T_1MP	L	48	1.1E-7	1.5E-3	1.7E-10	1.7E-10
109	T_2MP	L	51	1.3E-6	5.4E-4	7.0E-10	7.0E-10
110	T_3MP	L	51	9.2E-8	5.4E-4	5.0E-11	5.0E-11
111	T_4MP	L	56	7.8E-9	2.0E-3	1.6E-11	1.6E-11
112	T_5MP	L	61	3.9E-9	2.2E-3	8.6E-12	8.6E-12
					Sum =	8.09E-5	1.13E-4

Table B.22 St. Lucie Core Melt Accident Sequence Probabilities for Bleed and Feed with the PORVs Unblocked After Recovery

Anal. No.	Multi-plier	Boolean Sequence	SL Seq. No.	Multi. Freq.	Boolean Sequence Freq.	Core Melt Freq.	Core Melt Frequencies for Sequences Replaced
84	$S\ M$	$L\bar{D}\ \bar{P}\ H$	42				
85	S_2M	$L\bar{D}_1P_1$	43				
86	$T_1M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	44				
87	$T_1M\bar{P}Q$	$L\bar{D}_1P_1$	45				
88	$T_2M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	42				
89	$T_2M\bar{P}Q$	$L\bar{D}_1P_1$	43				
90	$T_3M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	42				
91	$T_3M\bar{P}Q$	$L\bar{D}_1P_1$	43				
92	T_3Q	$L\bar{D}_1\bar{D}_1H_1$	42				
93	T_3Q	$L\bar{D}_1P_1$	43				
94	$T_4M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	52				
95	$T_4M\bar{P}Q$	$L\bar{D}_1P_1$	53				
96	$T_5M\bar{P}Q$	$L\bar{D}_1\bar{P}_1H_1$	57				
97	$T_5M\bar{P}Q$	$L\bar{D}_1P_1$	58				
98	$T_1M\bar{P}\bar{Q}$	$L\bar{E}H_1$	46				
99	$T_1M\bar{P}\bar{Q}$	LE	47	8.3E-2	8.6E-5	7.1E-6	* 7.9E-6
100	$T_2M\bar{P}\bar{Q}$	$L\bar{E}H_1$	49				

0315

Table B.22 St. Lucie Core Melt Accident Sequence Probabilities for Bleed and Feed with the PORVs Unblocked After Recovery (cont.)

Anal. No.	Multi- plier	Boolean Sequence	SL Seq. No.	Multi. Freq.	Boolean Sequence Freq.	Core Melt Freq.	Core Melt Frequencies for Sequences Replaced
101	$T_2M\bar{P}\bar{Q}$	LE	50	9.7E-1	9.7E-8	9.4E-8	1.9E-7
102	$T_3M\bar{P}\bar{Q}$	$L\bar{E}H_1$	49				
103	$T_3M\bar{P}\bar{Q}$	LF	50				
104	$T_4M\bar{P}\bar{Q}$	$L\bar{E}H_1$	54				
105	$T_4M\bar{P}\bar{Q}$	LE	55	5.8E-3	1.4E-6	8.1E-9	1.7E-7
106	$T_5M\bar{P}\bar{Q}$	$L\bar{E}H_1$	59				
107	$T_5M\bar{P}\bar{Q}$	LE	60	2.9E-3	1.4E-5	4.1E-8	4.1E-8
108	T_1MP	L	48				
109	T_2MP	L	51				
110	T_3MP	L	51				
111	T_4MP	L	56				
112	T_5MP	L	61				
					Sum =	7.24E-6	8.30E-6

0316

Table B.23 St. Lucie Boolean Accident Sequences After Recovery Has Been Applied

```
                     SILR1 =

   1   2.4320E-05    SIS072B-VCC-LF * SIS072A-VCC-LF * RA-VCCH +
   2   [illegible]   RASL[illegible]-UTM-CM * RA-6H +
   3   1.1552E-06    SIS072A-VCC-LF * RA-VCCH * CCW14301-XOC-UTM +
   4   1.1552E-06    SIS072B-VCC-LF * RA-VCCH * CCW14255-XOC-UTM +
   5   1.1552E-06    SIS072B-VCC-LF * RA-VCCH * CCW14167-XOC-UTM +
   6   1.1552E-06    SIS072A-VCC-LF * RA-VCCH * CCW14178-XOC-UTM +
   7   1.1552E-06    SIS072A-VCC-LF * RA-VCCH * CCW14270-XOC-UTM +
   8   1.1552E-06    SIS072B-VCC-LF * RA-VCCH * CCW14289-XOC-UTM +
   9   9.1200E-07    SIS072A-VCC-LF * RA-VCCH * SIS072B-VCC-UTM +
  10   9.1200E-07    SIS072B-VCC-LF * RA-VCCH * SIS072A-VCC-UTM +
  11   5.7000E-07    ICW1A-PMS-LF * SIS7172-CCC-LF +
  12   5.7000E-07    CCW1A-PMS-LF * SIS7172-CCC-LF +
  13   5.7000E-07    CCW1B-PMS-LF * SIS7174-CCC-LF +
  14   3.0400E-07    SIS072A-VCC-LF * RA-VCCH * ICW21232-XOC-LF +
  15   3.0400E-07    SIS072A-VCC-LF * RA-VCCH * ICW21233-XOC-LF +
  16   3.0400E-07    SIS072A-VCC-LF * RA-VCCH * CCW14146-CCC-LF +
  17   3.0400E-07    SIS072A-VCC-LF * RA-VCCH * CCW14133-XOC-LF +
  18   3.0400E-07    SIS072A-VCC-LF * RA-VCCH * CCW14178-XOC-LF +
  19   3.0400E-07    SIS072A-VCC-LF * RA-VCCH * CCW14160-XOC-LF +
  20   3.0400E-07    SIS072B-VCC-LF * RA-VCCH * SIS7174-CCC-LF +


 THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS    2.43200000000E-05
      (THE SUM OF THE TERM VALUES IS    5.4966758799999E-05)
```

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0317

Table B.23 St. Lucie Boolean Accident Sequences After Recovery Has Been Applied (cont.)

```
                    STLRTA =
 1   4.0000E-05     CCW-PUMP-CM +
 2   1.0000E-05     ICW-PUMP-CM +
 3   6.0000E-06     SISLOG-UTM-CM * RA-6D +
 4   9.7470E-07     CCW1B-PMS-LF * ICW1A-PMS-LF * RA-18D +
 5   9.7470E-07     CCW1B-PMS-LF * CCW1A-PMS-LF * RA-18D +
 6   9.0000E-07     RA-6D * SIS-SENS-CM +
 7   8.5500E-07     CCW1B-PMS-LF * SISACT-A-LF * RA-6D +
 8   8.5500E-07     ICW1A-PMS-LF * SISACT-B-LF * RA-6D +
 9   8.5500E-07     CCW1A-PMS-LF * SISACT-B-LF * RA-6D +
10   7.5000E-07     SISACT-B-LF * SISACT-A-LF * RA-6D +
11   6.0000E-07     LOSS-OSP * DIESEL-CM * RA-1D * RA-10D +
12   5.7760E-07     LOSS-OSP * DGN2-GEN-LF * DGN1-GEN-LF * RA-1D * RA-11D +
13   4.3143E-07     CCW1B-PMS-LF * HPI1A-PMS-LF * LPI1A-PMS-LF +
14   3.2490E-07     ICW1A-PMS-LF * ICW1B-PMS-LF * ICW1C-PMS-UTM +
15   3.2490E-07     CCW1A-PMS-LF * ICW1B-PMS-LF * ICW1C-PMS-UTM +
16   2.8266E-07     ICW1A-PMS-LF * ICW1B-PMS-LF * ICW1C-PMS-LF +
17   2.8266E-07     CCW1A-PMS-LF * ICW1B-PMS-LF * ICW1C-PMS-LF +
18   2.2800E-07     CCWHX-1A-UTM * ICW21232-XOC-UTM +
19   2.2800E-07     CCWHX-1A-UTM * ICW144B-VOC-UTM +
20   2.2800E-07     CCWHX-1A-UTM * ICW21233-XOC-UTM +
21   2.2800E-07     CCWHX-1A-UTM * ICW21212-XOC-UTM +

 THE MAXIMUM TERM VALUE FOR COMPUTATION ? IS   4.000000000000E-05
      (THE SUM OF THE TERM VALUES IS   9.529417249993E-05)
```

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Table B.23 St. Lucie Boolean Accident Sequences After Recovery Has Been Applied (cont.)

```
                    STLR38 =

 1   4.0000E-05    CCW-PUMP-CM +
 2   1.0000E-05    ICW-PUMP-CM +
 3   2.0000E-06    SISLOG-UTM-CM * RA-6F +
 4   4.3143E-07    CCW1B-PMS-LF * HPI1A-PMS-LF * LPI1A-PMS-LF +
 5   4.0000E-07    LOSS-DSP * DIESEL-CM * RA-1F * RA-10F +
 6   3.8788E-07    LOSS-DSP * DGN2-GEN-LF * DGN1-GEN-LF * RA-1F * RA-11F +
 7   3.2490E-07    ICWIA-PMS-LF * ICWIB-PMS-LF * ICWIC-PMS-UTM +
 8   3.2490E-07    CCWIB-PMS-LF * CCWIA-PMS-LF * RA-18F +
 9   3.2490E-07    CCWIA-PMS-LF * ICWIB-PMS-LF * ICWIC-PMS-UTM +
10   3.2490E-07    CCWIB-PMS-LF * ICWIA-PMS-LF * RA-18F +
11   3.0000E-07    STS-SENS-CM * RA-6F +
12   2.8500E-07    CCWIB-PMS-LF * SISACT-A-LF * RA-5F +
13   2.8500E-07    ICWIA-PMS-LF * SISACT-B-LF * RA-5F +
14   2.8500E-07    CCWIA-PMS-LF * SISACT-B-LF * RA-6F +
15   2.8265E-07    CCWIA-PMS-LF * ICWIB-PMS-LF * ICWIC-PMS-LF +
16   2.8265E-07    ICWIA-PMS-LF * ICWIB-PMS-LF * ICWIC-PMS-LF +
17   2.5000E-07    SISACT-B-LF * SISACT-A-LF * RA-6F +
18   2.2800E-07    CCWHX-1A-UTM * ICW144B-VOC-UTM +
19   2.2800E-07    CCWHX-1A-UTM * ICW21233-XOC-UTM +
20   2.2800E-07    CCWHX-1A-UTM * ICW21212-XOC-UTM +

   THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS   4.0000000000000E-05
       (THE SUM OF THE TERM VALUES IS   8.325169349994E-05)
```

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0319

Table B.23 St. Lucie Boolean Accident Sequences After Recovery Has Been Applied (cont.)

```
                    STLR15 =

 1    5.1480E-05     RA-IF * RA-VCCF * BATTERY-CM +

 2    1.4175E-05     DIESEL-CM * RA-IIF * RA-IF * AFWIC-PTS-LF +

 3    1.3545E-05     RA-IF * DGN1-GEN-LF * DGN2-GEN-LF * RA-IIF * AFWIC-PTS-LF +

 4    8.3023E-05     RA-IF * DGN1-GEN-LF * BAT125-B-ZBT-LF * RA-VCCF +

 5    4.3705E-06     RA-IF * DGN1-GEN-LF * BAT125-B-ZBT-UTM * RA-VCCF +

 6    2.4705E-06     DIESEL-CM * RA-IIF * RA-IF * AFWIC-PTS-UTM +

 7    2.3783E-06     RA-IF * DGN1-GEN-LF * DGN2-GEN-LF * RA-IIF * AFWIC-PTS-UTM +

 8    2.1303E-06     RA-IF * DGN1-GEN-LF * RA-IIF * DGN2-GEN-UTM * AFWIC-PTS-LF +

 9    2.1303E-06     RA-IF * DGN2-GEN-LF * RA-IIF * DGN1-GEN-UTM * AFWIC-PTS-LF +

10    1.4042E-05     DIESEL-CM * RA-IF * RA-VCCF * AFW0803-VCC-LF +

11    1.3515E-06     RA-IF * DGN1-GEN-LF * DGN2-GEN-LF * RA-VCCF * AFW0803-VCC-LF +

12    1.2947E-06     RA-IF * DGN1-GEN-UTM * BAT125-B-ZBT-LF * RA-VCCF +

13    7.8507E-07     RA-IF * DGN1-GEN-UTM * BAT125-B-ZBT-UTM * RA-VCCF +

14    5.2274E-07     RA-IF * DGN2-GEN-LF * BAT125-A-ZBT-LF * RA-9F * AFWIC-PTS-LF +

15    3.9501E-07     RA-IF * DGN2-GEN-LF * RA-IIF * BAT125-A-ZBT-UTM * AFWIC-PTS-LF +

16    3.8177E-07     RA-IF * DGN2-GEN-LF * RA-IIF * DGN1-GEN-UTM * AFWIC-PTS-UTM +

17    3.8177E-07     RA-IF * DGN1-GEN-LF * RA-IIF * DGN2-GEN-UTM * AFWIC-PTS-UTM +


        THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS    5.1480000000000E-05
             (THE SUM OF THE TERM VALUES IS    1.1363510596205E-04)
```

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Table B.23 St. Lucie Boolean Accident Sequences After Recovery Has Been Applied (cont.)

```
                    STLR17 =

  1   3.3440E-05    BATTERY-CM * RA-IG * RA-VCCG +
  2   9.4500E-06    DIESEL-CM * AFWIC-PTS-LF * RA-IG * RA-IIG +
  3   9.3972E-05    DGN1-GEN-LF * DGN2-GEN-LF * AFWIC-PTS-LF * RA-IG * RA-IIG +
  4   5.1994E-06    DGN1-GEN-LF * BAT125-B-ZBT-LF * RA-IG * RA-VCCG +
  5   3.1753E-06    DGN1-GEN-LF * BAT125-B-ZBT-UTM * RA-IG * RA-VCCG +
  5   2.0823E-06    DGN2-GEN-LF * AFWIC-PTS-LF * AFWIA-PMS-LF * RA-IG * RA-IIG +
  7   2.0828E-05    DGN1-GEN-LF * AFWIC-PTS-LF * AFWIB-PMS-LF * RA-IG * RA-IIG +
  8   2.0799E-05    DGN1-GEN-LF * DGN2-GEN-LF * AFWCB03-VCC-LF * RA-IG * RA-IIG +
  9   1.6470E-06    DIESEL-CM * AFWIC-PTS-UTM * RA-IG * RA-IIG +
 10   1.0855E-06    DGN1-GEN-LF * DGN2-GEN-LF * AFWIC-PTS-UTM * RA-IG * RA-IIG +
 11   1.4653E-06    DGN1-GEN-LF * DGN2-GEN-UTM * AFWIC-PTS-LF * RA-IG * RA-IIG +
 12   1.4653E-06    DGN2-GEN-LF * DGN1-GEN-UTM * AFWIC-PTS-LF * RA-IG * RA-IIG +
 13   1.3583E-05    DGN1-GEN-LF * AFW-PUMP-CM * RA-IG * RA-IIG +
 14   1.3583E-05    DGN2-GEN-LF * AFW-PUMP-CM * RA-IG * RA-IIG +
 15   1.1392E-05    BAT125-B-ZBT-LF * AFWIA-PMS-LF * RA-IG * RA-VCCG +
 16   9.1200E-07    DIESEL-CM * AFWCB03-VCC-LF * RA-IG * RA-VCCG +
 17   8.5449E-07    DGN1-GEN-UTM * BAT125-B-ZBT-LF * RA-IG * RA-VCCG +

        THE MAXIMUM TERM VALUE FOR COMPUTATION 1 IS    3.3440000000000E-05
              (THE SUM OF THE TERM VALUES IS    9.5375021259996E-05)
```

0321

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Table B.24 St. Lucie Dominant Cut Sets by Accident Sequence

Anal. No.	Boolean Event #	Core Melt Failure Sequence	Cut Set Description	Cut Set Core Melt Probability
9	17	T_1MLE	BATTERY-CM	2.8E-6
			DIESEL-CM*AFW1C-PTS-LF	7.8E-7
			DGN1-GEN-LF*DGN2-GEN-LF*AFWalC-PTS-LF	7.6E-7
			DGN1-GEN-LF*BAT125-B-ZBT-LF	4.3E-7
			DGN1-GEN-LF*BAT125-B-ZBT-UTM	2.6E-7
			All other cut sets	2.9E-6
3	3a	$S_2MD_1D_2$	CCW-PUMP-CM	8.0E-7
			ICW-PUMP-CM	2.0E-7
			SISLOG-UTM-CM	1.2E-7
			All other cut sets	7.9E-7
40	3b	$T_3QD_1D_2$	CCW-PUMP-CM	6.0E-7
			ICW-PUMP-CM	1.5E-7
			SISLOG-UTM-CM	3.0E-8
			All other cut sets	4.7E-7
1	1	$S_2MH_1'H_2'$	SIS072B-VCC-LF*SIS072A-VCC-LF	4.9E-7
			RASLOG-UTM-CM	4.0E-8
			All other cut sets	5.7E-7
38	1	$T_3QH_1'H_2'$	SIS072B-VCC-LF*SIS072A-VCC-LF	3.6E-7
			RASLOG-UTM-CM	3.0E-8
			All other cut sets	4.3E-7
23	15	T_1MQLD_1	BATTERY-CM	1.3E-7
			DIESEL-CM*AFW1C-PTS-LF	3.7E-8
			DGN1-GEN-LF*DGN2-GEN-LF*AFW1C-PTS-LF	3.5E-8
			DGN1-GEN-LF*BAT125-B-ZBT-LF	2.1E-8
			DGN1-GEN-LF*BAT125-B-ZBT-UTM	1.3E-8
			All other cut sets	5.6E-8

The next step is to sort all the most significant cut sets identified for the top 6 accident sequences, and sum the total core melt probability for each cut set over all the accident sequences to which it contributes. This is given in Table B.25. These cut sets are the vulnerabilities described in Section 4 of the main report and subsequently communicated to the AE and discussed at the second plant visit along with those vulnerabilities identified for the special emergencies.

The internal analysis started with 64 accident sequences which included both secondary blowdown and bleed and feed capability. These 64 sequences had a total core melt probability of 1.81E-4 before recovery or technical specification violations were considered. After recovery was applied to 39 of the dominant sequences, the total core melt probability was reduced to 1.44E-5. The dominant cut sets accounted for 8.06E-6 of this total probability after recovery. In order to calculate the residual core melt probability that is not being addressed, consider the following:

1) The initial core melt probability before recovery was applied to the 64 accident sequences was 1.8067E-4.

2) There were 25 accident sequences not significant enough to which to apply recovery factors. This is a conservative estimate of the first residual core melt probability. 2.16E-6

3) The 39 accident sequences to which recovery was applied had a total core melt probability of 1.785E-4.

4) The core melt probability of these 39 accident sequences after recovery was 1.436E-5.

5) 33 of these accident sequences are not considered further. This is the second residual core melt probability. 1.14E-6

6) The top 6 accident sequences have dominant cut sets which became the basis of the potential vulnerabilities identified and analyzed in Section 5.3 The other cut sets in these 6 accident sequences are the third residual core melt probability. This is very conservative since modifications will also improve the non-dominant cut sets. 5.22E-6

7) Thus the total (conservative) residual core melt probability that we did not address in this phase of this internal analysis is the sum of these three residuals. 8.52E-6

0323

Table B.25 St. Lucie Dominant Cut Sets

	Dominant Cut Sets	Total Core Melt Probability Contribution
1.	BATTERY-CM	2.9E-6
2.	CCW-PUMP-CM	1.4E-6
3.	SIS072B-VCC-LF*SIS-072A-VCC-LF	8.5E-7
4.	DIESEL-CM*AFW1C-PTS-LF	8.2E-7
5.	DGN1-GEN-LF*DGN2-GEN-LF*AFW1C-PTS-LF	8.0E-7
6.	DGN1-GEN-LF*BAT125-B-ZBT-LF	4.5E-7
7.	ICW-PUMP-CM	3.5E-7
8.	DGN1-GEN-LF*BAT125-B-ZBT-UTM	2.7E-7
9.	SISLOG-UTM-CM	1.5E-7
10.	RASLOG-UTM-CM	7.0E-8

The value analysis of the modifications for the 10 vulnerabilities identified (corresponding to the 10 groups of cut sets shown in Table B.25) determines the actual improvement in core melt probability.

5.4 Containment Systems Integration

In order to determine public risk measures, the core melt accident sequences must be expanded to include containment systems and then analyzed for containment failure modes, assigned release categories, and propagated in a consequence analysis. Therefore, the next step was to apply the containment event tree to each of the 12 dominant accident sequences from Table B.12.

The containment systems event tree is shown in Figure B.5 in a previous subsection of this Appendix. Each of the 12 dominant core melt accident sequences had to be rerun with each containment system sequence. The outcomes of the containment systems event tree cover all the combinations of success or failure of containment overpressure protection (COP) and post accident radioactivity removal (PARR). It should be noted that even the containment system success sequence $\bar{Y}_1\bar{C}_1\bar{F}'$ is important since the containment may fail by some means not identified by the containment systems event tree given a core melt.

The matrix of the SETS runs is given in Table B.26. Each distinctly different Boolean accident sequence (there are 8) corresponding to the 12 dominant accident sequences was run with all the Boolean containment system sequences. This is necessary due to the dependencies between systems. For example, consider the analysis sequence number 9 which corresponds to TP sequence number 17. The multiplier expression is $\dot{T}_1M\bar{P}\bar{Q}$ and the Boolean events is LE. Thus LE must be run with each containment sequence, i.e.,

$LE\bar{Y}_1\bar{C}_1\bar{F}'$

$LE\bar{Y}_1\bar{C}_1F'$

$LE\bar{Y}_1C_1$

$LEY_1\bar{C}_1\bar{F}'$

$LEY_1\bar{C}_1F'$, and

LEY_1C_1.

Table B.26 St. Lucie Containment Sequences

	1	2	3	4	5	6
SL No.	$\bar{Y}_1\bar{C}_1\bar{F}'$	$\bar{Y}_1\bar{C}_1F'$	$\bar{Y}_1C_1$	$Y_1\bar{C}_1\bar{F}'$	$Y_1\bar{C}_1F'$	Y_1C_1
1	*	*	*	*	*	*
2						
3	*	*	*	*	*	*
4						
5						
6						
7						
8						
9						
10						
11	*	*	*	*	*	*
12						
13						
14						
15	*	*	*	*	*	*
16						
17	*	*	*	*	*	*
18						
19						
20	*	*	*	*	*	*
21						
22						
23						
24						
25						
26						
27						

0326

Table B.26 St. Lucie Containment Sequences (cont.)

	1	2	3	4	5	6
SL No.	$\bar{Y}_1\bar{C}_1\bar{F}'$	$\bar{Y}_1\bar{C}_1F'$	$\bar{Y}_1C_1$	$Y_1\bar{C}_1\bar{F}'$	$Y_1\bar{C}_1F'$	Y_1C_1
28						
29						
30	*	*	*	*	*	*
31						
32						
33						
34						
35						
36						
37						
38						
39						
40	*	*	*	*	*	*
41						
42						
43						
44						
45						
46						
47						
48						

0327

The corresponding probabilities would be $p(T_1M\bar{P}\bar{Q}) * p(LE\bar{Y}_1\bar{C}_1\bar{F}')$, etc. It would by incorrect to assume that

$$p(LE\bar{Y}_1\bar{C}_1\bar{F}') = p(LE)p(\bar{Y}_1\bar{C}_1\bar{F}').$$

After the SETS runs were completed the resulting probabilities for the total accident sequence were screened to determine those sequences to be carried on to the recovery analysis and placed into release categories. The results before recovery are given in Table B.27 and after recovery in Table B.28. The next step in the analysis leading to public risk measures can be found in Section 7.2 of the main report.

6.0 VALUE ANALYSIS

6.1 Core Melt Probabilities

The original purpose of this section was to determine and present the estimated core melt probabilities for the alternatives selected to address the potential vulnerabilities identified in Section 5.3 of this appendix. During the course of the analysis, however, it was discovered that no individual vulnerability contributes significantly enough to the total core melt probability to warrant a proposed modification. Section 2.3 of the main report does use engineering judgment, though, to recommend a few modifications to reduce the need for manual recovery of the vulnerabilities that do exist.

In the value analysis it is the change in core melt probability that results from an alternative that is most important. For St. Lucie, no modifications are proposed to handle the vulnerabilities of Table B.25 with the exception of an add-on decay heat removal train in Alternative 2. Nevertheless, it is informative to note the anticipated reasons and value of each recommended modification. This is done in Section 6.1.1.

6.1.1 Modification Values

Of the ten vulnerabilities listed in Table B.25, none is significant enough to warrant a proposed modification at this time. This is due mainly to the ability of plant personnel to manually recover from failed actuation signals, a tripped power conversion system, failed motor operated valve control circuits, etc. Only the intake cooling water pump common mode (#7) failure and component cooling water pump common mode (#2) failures are considered nonrecoverable. The terms that denote battery common mode failure (#1), diesel generator common mode failure (#4), battery local fault (#6), and diesel generator local faults (#5, 6, 8) all have some probability for manual fixing that reduces the core melt contribution based upon the time available until core uncovery. Similarly, the motor operated sump valves (#3) could be manually opened and the

Table B.27 Total Accident Sequence Probabilities Including Containment Systems Failures Before Recovery Considered - Base Case

Core Melt Accident Sequence	Anal. No.	SL Boolean Sequence Number	Core Melt Failure Sequence	Containment Systems Sequences					
				$\bar{Y}_1\bar{C}_1\bar{F}'$	$\bar{Y}_1\bar{C}_1F'$	$\bar{Y}_1C_1$	$Y_1\bar{C}_1\bar{F}'$	$Y_1\bar{C}_1F'$	Y_1C_1
$S_2M\bar{L}\bar{D}_1H_1'H_2'$	1	1	$S_2MH_1'H_2'$	2.6E-8	1.9E-5	1.5E-7	Φ	Φ	Φ
$S_2M\bar{X}\bar{L}D_1D_2$	3	3	$S_2MD_1D_2$	3.4E-8	1.7E-8	9.1E-6	Φ	Φ	4.9E-6
$T_1M\bar{P}\bar{Q}LE$	9	17	T_1MLE	2.1E-5	4.0E-8	5.8E-8	Φ	3.1E-8	6.4E-5
$T_2M\bar{P}\bar{Q}LE$	12	20	T_2MLE	8.6E-6	Φ	Φ	Φ	Φ	4.3E-7
$T_1MQ\bar{X}\bar{L}D_1D_2$	19	11	$T_1MQD_1D_2$	2.0E-9	Φ	3.5E-8	Φ	1.4E-9	1.1E-6
$T_1M\bar{P}QLD_1$	23	15	T_1MQLD_1	9.4E-10	Φ	Φ	Φ	9.7E-10	2.0E-6
$T_2MQ\bar{L}\bar{D}_1H_1'H_2'$	24	1	$T_2MQH_1'H_2'$	2.7E-9	2.0E-6	1.6E-8	Φ	Φ	Φ
$T_2MQ\bar{X}\bar{L}D_1D_2$	26	3	$T_2MQD_1D_2$	3.6E-9	1.8E-9	9.5E-7	Φ	Φ	5.2E-7
$T_3Q\bar{L}\bar{D}_1H_1'H_2'$	38	1	$T_3QH_1'H_2'$	1.9E-8	1.4E-5	1.1E-7	Φ	Φ	Φ
$T_3Q\bar{X}\bar{L}D_1D_2$	40	3	$T_3QD_1D_2$	2.5E-8	1.3E-8	6.8E-6	Φ	Φ	3.7E-6
$T_4M\bar{P}\bar{Q}LE$	53	30	T_4MLE	1.2E-5	2.1E-7	3.0E-7	2.6E-9	2.5E-8	1.5E-7
$T_5M\bar{P}\bar{Q}LE$	63	40	T_5MLE	6.1E-6	9.0E-10	9.0E-10	Φ	Φ	1.4E-7
			TOTAL	4.78E-5	3.53E-5	1.75E-5	2.6E-9	5.84E-8	7.69E-5

0329

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Table B.28 Total Accident Sequence Probabilities Including Containment Systems Failures After Recovery Considered - Base Case

Core Melt Accident Sequence	Anal. No.	SL Boolean Sequence Number	Core Melt Failure Sequence	Containment Systems Sequences					
				$\bar{Y}_1\bar{C}_1\bar{F}'$	$\bar{Y}_1\bar{C}_1F'$	$\bar{Y}_1C_1$	$Y_1\bar{C}_1\bar{F}'$	$Y_1\bar{C}_1F'$	Y_1C_1
$S_2M\bar{L}\bar{D}_1H_1'H_2'$	1	1	$S_2MH_1'H_2'$		1.0E-6				
$S_2M\bar{X}\bar{L}D_1D_2$	3	3	$S_2MD_1D_2$			4.2E-7			1.2E-6
$T_1M\bar{P}\bar{Q}LE$	9	17	T_1MLE	1.5E-6				4.3E-9	5.9E-6
$T_2M\bar{P}\bar{Q}LE$	12	20	T_2MLE	1.2E-7					
$T_1MQ\bar{X}\bar{L}D_1D_2$	19	11	$T_1MQD_1D_2$						1.9E-7
$T_1M\bar{P}QLD_1$	23	15	T_1MQLD_1						2.9E-7
$T_2MQ\bar{L}\bar{D}_1H_1'H_2'$	24	1	$T_2MQH_1'H_2'$		1.0E-7				
$T_2MQ\bar{X}\bar{L}D_1D_2$	26	3	$T_2MQD_1D_2$			2.5E-8			
$T_3Q\bar{L}\bar{D}_1H_1'H_2'$	38	1	$T_3QH_1'H_2'$		7.5E-7				
$T_3Q\bar{X}\bar{L}D_1D_2$	40	3	$T_3QD_1D_2$			1.8E-7			8.5E-7
$T_4M\bar{P}\bar{Q}LE$	53	30	T_4MLE	1.7E-7			5.4E-11	6.6E-10	
$T_5M\bar{P}\bar{Q}LE$	63	40	T_5MLE	1.9E-8					
			TOTAL	1.81E-6	1.85E-6	6.25E-7	5.40E-11	4.96E-9	8.43E-6

safety injection signal (#9) and recirculation actuation signals (#10) could be manually initiated.

However, it may be prudent to make a few minor modifications to reduce the reliance on human recovery in possibly short time periods. One such modification would involve transferring DC power of AFW MV-08-14 and MV-09-12 to 125V DC bus 1A. This would increase system dependency on the single DC bus 1B and decrease the core melt contributions of internal vulnerabilities #6 and #8 by approximately 2.3E-3 each.

By allowing for manual cranking of the diesel generators or a redundant or dedicated diesel generator battery starter, the core melt contribution from vulnerability #1 could be reduced by factors of 1E-2 to 2E-3, respectively. Providing for rapid availability of Unit 2 AC electric power would reduce the contributions of vulnerabilities #4, 5, 6, and 8 by approximately 1E-2 each.

6.1.2 Alternative Values

In this St. Lucie analysis, no modifications are being proposed to address the individual internal vulnerabilities presented in Table B.25. Instead, the value of an alternative decay heat removal train is calculated here (alternative 2) as well as in Section 7 of the main report. The inputs to that calculation from the internal analysis are the core melt probability determined in Section 5.2 of this appendix and the reduced core melt probability if this add-on is implemented.

Base Core Melt Probability - The base core melt probability with credit for bleed and feed and secondary blowdown capability is given in Section 5.2 of this appendix as 1.44E-5. This probability includes all of the recovery actions, technical specification considerations, and non-dominant sequences and cut sets. Table B.29 represents the 12 dominant sequences from Table B.12 after recovery with a total core melt probability of 1.41E-5.

Alternative 1 Core Melt Probability - Since the core melt probability for St. Lucie is dominated by external events, alternative 1 proposes modifications for these scenarios. These sequences are not part of the internal analysis and are quantified in Chapters 5 and 7.

Alternative 2 Core Melt Probability - This alternative is a dedicated shutdown heat removal add-on train which, because of its independence, affects every sequence. When the system responds to a sequence which includes a loss of offsite power, it uses its own electric power supply. At this time the probability of system unavailability is approximately 9.8E-2. If offsite AC power is available the add-on system can use it and the system unavailability is 5.2E-2. Table B.30 presents

0331

Table B.29 St. Lucie Results of Top 12 Sequences After Recovery - Base Case

Analysis Number	Core Melt Failure Sequence	Core Melt Frequency	SL Boolean Event Sequence Number
9	T_1MLE	7.9E-6	17
3	$S_2MD_1D_2$	1.9E-6	3
40	$T_3QD_1D_2$	1.2E-6	3
1	$S_2MH_1'H_2'$	1.1E-6	1
38	$T_3QH_1'H_2'$	8.3E-7	1
23	T_1MQLD_1	2.9E-7	15
19	$T_1MQD_1D_2$	1.9E-7	11
12	T_2MLE	1.9E-7	20
26	$T_2MQD_1D_2$	1.7E-7	3
53	T_4MLE	1.7E-7	30
24	$T_2MQH_1'H_2'$	1.2E-7	1
63	T_5MLE	4.1E-8	40

Sum = 1.41E-5

Table B.30 St. Lucie Results of Top 12 Sequences After Recovery - Alternative 2

Analysis Number	Core Melt Failure Sequence	Core Melt Frequency	SL Boolean Event Sequence Number
9	T_1MLE	7.7E-7	17
3	$S_2MD_1D_2$	9.9E-8	3
40	$T_3QD_1D_2$	6.2E-8	3
1	$S_2MH_1'H_2'$	5.7E-8	1
38	$T_3QH_1'H_2'$	4.3E-8	1
23	T_1MQLD_1	2.8E-8	15
19	$T_1MQD_1D_2$	1.9E-8	11
12	T_2MLE	9.9E-9	20
26	$T_2MQD_1D_2$	8.8E-9	3
53	T_4MLE	8.8E-9	30
24	$T_2MQH_1'H_2'$	6.2E-9	1
63	T_5MLE	2.1E-9	40
		Sum = 1.11E-6	

the core melt probabilities of the top 12 internal event sequences after alternative 2 (with recovery) has been implemented. The core melt probability is reduced from 1.41E-5 to 1.11E-6 by adding this dedicated system.

6.2 Containment System Probabilities

The same procedure done in Section 5.4 of this appendix was repeated for alternative 2. The end result is Table B.31 which gives the probabilities of the 12 most significant core melt sequences combined with the containment system sequences after recovery.

Although the add-on dedicated SDHR system was not part of the internal analysis, Table B.31 is derived from the same information by applying the add-on probability factor. Table B.31 is generated by multiplying each entry of Table B.28 by the add-on factor (9.8E-2 if offsite power is unavailable and 5.2E-2 if offsite power is available). The T_1 sequences are the cases where offsite power is unavailable. This, then, provides the internal analysis calculations needed in Section 7.2 of the main report.

0334

Table B.31 Total Accident Sequence Probabilities Including Containment Systems Failures After Recovery Considered - Alternative 2

Core Melt Accident Sequence	Anal. No.	SL Boolean Sequence Number	Core Melt Failure Sequence	Containment Systems Sequences					
				$\bar{Y}_1\bar{C}_1\bar{F}'$	$\bar{Y}_1\bar{C}_1F'$	$\bar{Y}_1C_1$	$Y_1\bar{C}_1\bar{F}'$	$Y_1\bar{C}_1F'$	Y_1C_1
$S_2M\bar{L}\bar{D}_1H_1'H_2'$	1	1	$S_2MH_1'H_2'$		5.2E-8				
$S_2M\bar{X}\bar{L}D_1D_2$	3	3	$S_2MD_1D_2$			2.2E-8			6.2E-8
$T_1M\bar{P}\bar{Q}LE$	9	17	T_1MLE	1.5E-7				4.2E-10	5.8E-7
$T_2M\bar{P}\bar{Q}LE$	12	20	T_2MLE	6.2E-9					
$T_1MQ\bar{X}\bar{L}D_1D_2$	19	11	$T_1MQD_1D_2$						1.9E-8
$T_1M\bar{P}QLD_1$	23	15	T_1MQLD_1						2.8E-8
$T_2MQ\bar{L}\bar{D}_1H_1'H_2'$	24	1	$T_2MQH_1'H_2'$		5.2E-9				
$T_2MQ\bar{X}\bar{L}D_1D_2$	26	3	$T_2MQD_1D_2$			1.3E-9			
$T_3Q\bar{L}\bar{D}_1H_1'H_2'$	38	1	$T_3QH_1'H_2'$		3.9E-8				
$T_3Q\bar{X}\bar{L}D_1D_2$	40	3	$T_3QD_1D_2$			9.4E-9			4.4E-8
$T_4M\bar{P}\bar{Q}LE$	53	30	T_4MLE	8.8E-9			2.8E-12	3.4E-11	
$T_5M\bar{P}\bar{Q}LE$	63	40	T_5MLE	9.9E-10					
			TOTAL	1.66E-7	9.62E-8	3.27E-8	2.80E-12	4.54E-10	7.33E-7

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APPENDIX C

SEISMIC RISK ASSESSMENT OF THE ST. LUCIE NUCLEAR POWER PLANT

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0337

TABLE OF CONTENTS

		Page	
1.0	INTRODUCTION AND OVERVIEW	C-3	339
2.0	PLANT DESCRIPTION	C-8	344
3.0	DESCRIPTION OF PLANT SYSTEMS	C-9	345
4.0	SEISMICITY AND HAZARD CURVES	C-18	354
5.0	RESPONSE CALCULATIONS	C-26	362
6.0	SEISMIC FRAGILITIES	C-76	412
	6.1 Generic	C-76	412
	6.2 Site-Specific	C-79	415
7.0	CORE DAMAGE AND RISK COMPUTATIONS	C-82	418
	7.1 Event Trees	C-82	418
	7.2 Initiating Events	C-85	421
	7.3 Base Case Accident Sequence Probabilities	C-86	422
	7.4 Release Category Assignments	C-88	424
8.0	POTENTIAL DESIGN CHANGES AND RISK IMPACT	C-97	433
9.0	CONCLUSIONS	C-101	437
10.0	REFERENCES	C-102	438
ATTACHMENT C.1 -	Generic Fragilities for SSE = 0.10g	C-104	440
ATTACHMENT C.2 -	Review of Site-Specific Fragilities Derived for 19 Recent PRAs	C-120	456

1.0 INTRODUCTION

The US Nuclear Regulatory Commission has designated decay heat removal an Unresolved Safety Issue and established a program under Task Action Plan (TAP) A-45, Decay Heat Removal Requirements,(1) to provide the technical basis for resolution of the issue. As part of the program to establish a technical base for decision-making, probabilistic risk assessments are being performed for a number of example plants. These risk assessments are focused on the plant's response to small loss of coolant accidents (SLOCAs) and transient events which require the various decay heat removal systems to bring the plant to a cold shutdown state. As part of this risk assessment program, simplified seismic risk assessments are being performed for the plants under study. This report describes the seismic risk assessment for the St. Lucie Nuclear Power Plant.

In preparation for this work, a set of procedures for performing a simplified seismic risk assessment was developed (Reference 2). The result was a methodology that can be applied to a particular plant in a 16-week timeframe and at a cost of about $40K per plant. The simplifications utilized in this methodology were derived from the extensive results obtained in the NRC-sponsored Seismic Safety Margins Research Program (Reference 3). The simplified methodology is summarized below.

1.1 Summary of Seismic Risk Analysis Procedures

There are seven steps required for calculating the seismic risk at a nuclear power plant:

1. Determine the local earthquake hazard (Hazard Curve and Site Spectra).
2. Identify accident scenarios for the plant which lead to radioactive release. (Initiating Events and Event Trees).
3. Determine failure modes for the plant safety and support systems. (Fault Trees).
4. Determine fragilities (probabilistic failure criteria) for the important structures and components.
5. Determine the responses (accelerations or forces) of all structures and components (for each earthquake level).
6. Compute the probability of core melt using the information from Steps 1 through 5.
7. Estimate uncertainty in the core melt frequencies.

Only the level of detail differentiates a simplified seismic analysis from a detailed seismic PRA. The seven steps of the TAP A-45 simplified seismic risk analysis procedure are described below.

Step 1 - Seismic Hazard Characterization

(a) If a seismic hazard curve exists for the plant (e.g., from the Systematic Evaluation Program (SEP) program or from an existing PRA of a nearby plant), it is used.

(b) If no such curve exists, a hazard curve scaled to an exceedance probability of 2.5E-4/year at the safe shutdown earthquake (SSE) would be used. The slope of the curve for higher peak ground acceleration (pga) values is estimated from hazard curves for other sites in the same broad seismological province.

(c) Analyses of all plants west of the Rocky Mountains require site specific hazard curves due to the high levels of seismic activity, but most existing plants have hazard curves already available.

Step 2 - Initiating Events and Event Trees

The scope of TAP A-45 includes only the initiating events associated with small loss of coolant accident (SLOCA) and transient events. Two types of transients are being considered: those in which the power conversion system (PCS) is initially available (denoted Type T1 transients) and those in which the PCS is failed as a direct consequence of the initiating event (denoted Type T2 transients). The event trees derived for the internal event analyses for TAP A-45 are utilized.

The small LOCA initiating event frequencies are computed based on a statistical distribution of small pipe failures developed in the NRC-sponsored Seismic Safety Margins Research Program (SSMRP) program (Reference 3).

The frequency of Type T2 transients is based on the probability of loss of offsite power (LOSP). This is the dominant cause of this type of transient (for the majority of plants for which LOSP causes loss of main feedwater). The frequency of the Type T1 initiating event is computed from the condition that the sum of the initiating event probabilities must be unity. The hypothesis is that, given an earthquake of reasonable size, at least one of the initiating events will occur.

0340

Step 3 - Fault Trees

Simplified systems fault trees for a number of example plants have been developed as part of TAP A-45, (for random failures only). These fault trees will be used, although they require some modification to include basic events for seismic failure modes and re-solving the trees for pertinent cut sets to be included in the probabilistic calculations. Probabilistic culling is used in re-solving these trees to assure that important correlated failure modes are not lost.

Step 4 - Component and Structure Failure Descriptions

Component seismic fragilities are obtained both from a generic fragility data base and from plant-specific fragilities developed for components identified during the plant walk-down.

The generic data base of fragility functions for seismically-induced failures was originally developed as part of the SSMRP (Reference 4). Fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data utilized in developing fragility curves were obtained from the results of component manufacturer's qualification tests, independent testing lab failure data and data obtained from the extensive U.S. Corps of Engineers SAFEGUARD Sub-system Hardness Assurance Program. These data were statistically combined with the expert opinion survey data to produce fragility curves for each of the generic component categories.

Within the 16-week/plant time scale, structural fragility development is not possible, and is not being included in TAP A-45. This is appropriate since it is the design requirements for decay heat removal systems which are being evaluated, and not the structural design. However, the simplified seismic analysis approach being applied should prove widely useful in examining other generic and value/impact issues not directly concerned with adequacy of power plant structures.

Step 5 - Seismic Response of Structures and Components

Building and component seismic responses are estimated from peak ground accelerations at several probability intervals on the hazard curve. Three basic aspects of seismic response--best estimates, variability, and correlation--are generated. SSMRP Zion analysis results and simplified methods studies form the basis for assigning scaling, variability and correlation of responses.

In each case, SHAKE code (Reference 5) calculations are performed to assess the effect of the local soil column (if any) on the surface peak ground acceleration and soil structure

0341

interactions. This permits an evaluation of the effects of non-homogeneous underlying soil conditions which can strongly affect the building responses.

Two situations are commonly encountered in developing structural responses. First, for many early plants, only design floor spectra are available. For this situation, the results of a detailed soil-structure interaction investigation (Reference 6) made as part of the SSMRP may be used to scale the computed design floor slab accelerations to best estimate values including the effects of the soil, and to generate best estimate floor response spectra.

Second, one has the situation where fixed-base mass-spring or eigen-system descriptions are available for one or more buildings at the site. For this case, one can compute the floor slab accelerations using the CLASSI code (Reference 3). This code takes a fixed-base eigen-system model of the structure and input-specified frequency dependent soil impedances and computes the structural response (as well as variation in structural response if desired).

Variability in responses (floor and spectral accelerations) is assigned based on the SSMRP results. The recommended uncertainties (expressed as standard deviations of the logarithms of the responses, β) are shown below:

Quantity	β random
Peak Ground Acceleration	0.25
Floor Zero Period Acceleration	0.35
Floor Spectral Acceleration	0.45

Correlation between component failures is being included explicitly in TAP A-45. In computing the correlation between component failures (in order to quantify the cut-sets) it is necessary to consider correlations both in the responses and in the fragilities of each component. Inasmuch as there are no data as yet on correlation between fragilities, the fragility correlations between like components are taken as zero and 1, and the possible effect of such correlation quantified in a sensitivity study. The correlation between responses is assigned according to a set of rules. These are applied to both BWR and PWR plants.

Step 6 - Probabilistic Failure and Core Melt Calculations

Given the input from the five steps above, the SETS (Reference 7) code is used to calculate the required output (probabilities of failure, core melt, etc.). Use of the SEISIM code for final computation of accident sequences permits proper inclusion of seismically-induced correlation between component failures.

0342

Step 7 - Estimate Uncertainties

Uncertainties in core damage estimates are being based entirely on upper and lower confidence limits on the hazard curve. Generic recommendations for such limits are given in Reference 2. This simplification is appropriate since previous seismic PRAs have shown the hazard curve uncertainty to contribute 70-90% of the total seismic risk uncertainty.

2.0 PLANT DESCRIPTION

The St. Lucie nuclear power plant is a two-unit pressurized water reactor located midway down the east coast of the Florida peninsula. It is owned and operated by Florida Power and Light Company, and each unit is designed to produce 876 MW of electrical power. The architect-engineering firm was Ebasco, Inc. Unit 1 achieved commercial operation in December 1976, and Unit 2 achieved commercial operation in May 1983.

The plant utilizes two Combustion Engineering closed-cycle, 2-loop pressurized water nuclear steam supply systems. Cooling water is drawn from the Atlantic Ocean.

The plant is built on a deep soil site consisting of a layer of engineered fill approximately 78 feet thick over a 540 foot layer of natural sand and clay-sand deposits, which in turn overlay a limestone bedrock. Local site amplification of ground motion due to the soil layers was explicitly considered in defining the design earthquakes. Based on a study by Dr. R. B. Barksdale of the Georgia Institute of Technology (Ref. 8), the operating basis earthquake was chosen as 0.05g horizontal and 0.033g vertical. The safe shutdown earthquake (SSE) was defined as 0.10g horizontal.

0344

3.0 DESCRIPTION OF PLANT SYSTEMS

The important systems on the transient and small LOCA event trees are the Electric Power System, the Main Feedwater System, the Auxiliary Feedwater System, the High Pressure Injection System, Low Pressure Injection System and the Pilot Operated Relief Valves. In addition, certain operational functions such as Feed and Bleed Cooling and Secondary Blowdown are identified on the event trees. In the following, we briefly discuss these systems, and highlight the functional interactions which cause the dominant seismic failure modes discussed in Chapter 6.

Electric Power System

The schematic for this system is shown in Figure 3.1. Power from the switchyard is stepped down through the start-up transformers 1A and 1B to four 4160V busses (1A-2, 1A-3, 1B-2, 1B-3). Power for the safety injection pumps is taken directly off these busses. Two diesel generators (DG1A, DG1B) feed power to the 4160V busses if loss of offsite power (LOSP) occurs.

The power is further stepped down to 480V by the station transformers, and fed to five 480V load centers (1A-1, 1A-2, 1AB, 1B-1, 1B-2). The power is then fed to five 480V motor control centers (1A-5, 1A-6, 1AB, 1B-5, 1B-6) from which all additional safeguards AC power loads are taken.

In particular, the 480V busses feed the five DC battery chargers (1A, 1AA, 1AB, 1B, 1BB) which charge two station emergency battery banks. Thus, DC power is provided to three DC busses (A, AB, B) from the battery chargers (if 480AC is available) or from the batteries if AC power is not available. Note in Figure 3.1, however, that in its normal configuration, DC bus AB is energized only from DC bus B or via AC power from diesel 1B. This normal alignment plays an important role in failure modes of the AFWS as discussed below.

There are two independent trains of AC and DC power (Trains A and B), each train being fed by one of the diesel generators. Normally, all train A loads (pumps, valves, etc.) are fed from the train A power system, and train B loads from the B power system. The most important interaction here is that the diesel generators require DC power for start-up and control, and thus if no 480V power is available to the battery chargers, failure of the train A battery causes failure of the train A diesel generator, and the same for train B. Loss of both batteries in conjunction with complete LOSP results in complete station blackout.

Main Feedwater System

The main feedwater system is assumed to be isolated given that loss of off-site power has occurred. Because of the low median

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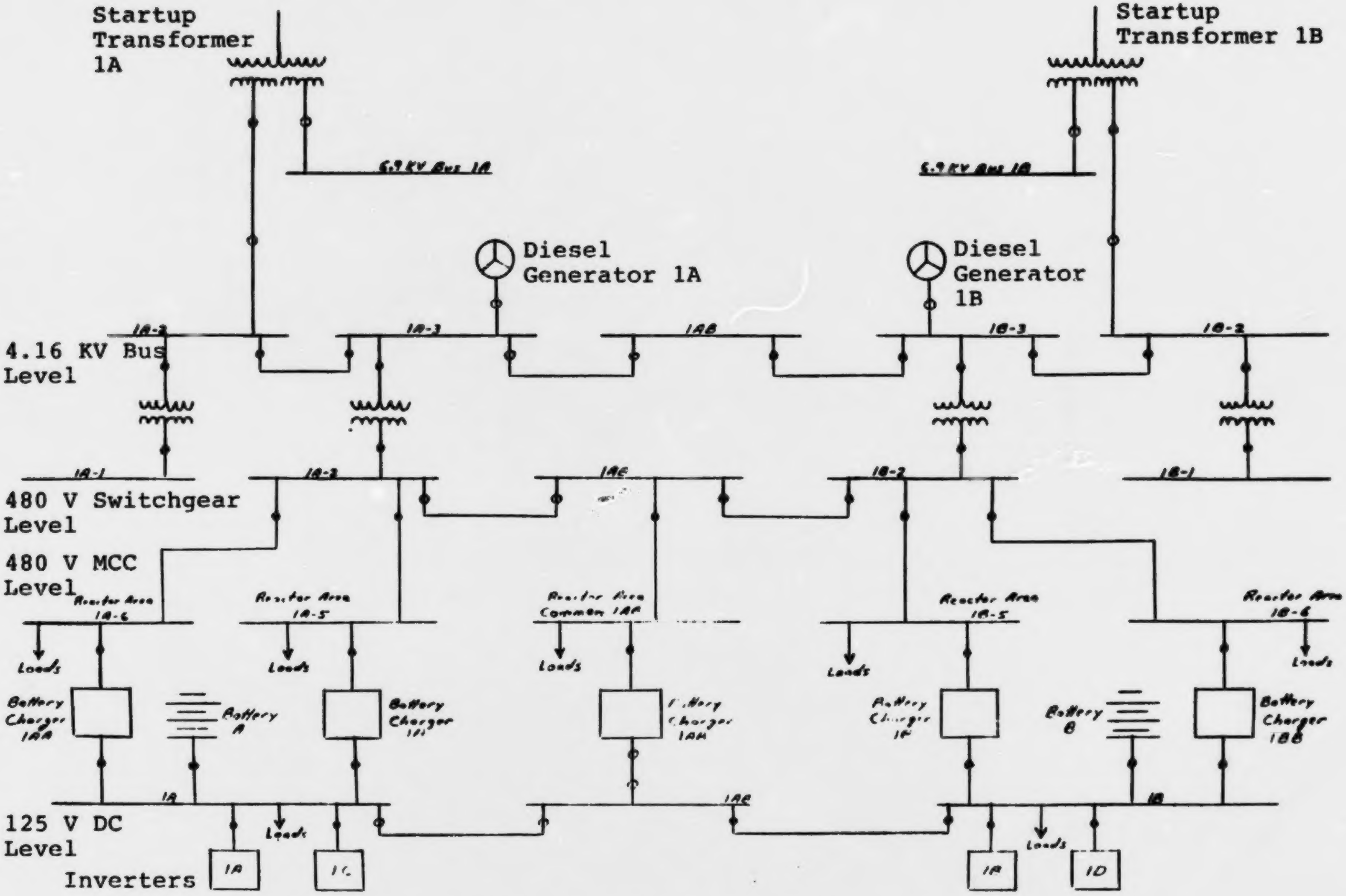
Startup Transformer 1A
Startup Transformer 1B
6.9 KV Bus 1A
6.9 KV Bus 1B
Diesel Generator 1A
Diesel Generator 1B
4.16 KV Bus Level
1A-2
1A-3
1AB
1B-3
1B-2
480 V Switchgear Level
1A-1
1A-2
1AB
1B-2
1B-1
480 V MCC Level
Reactor Area 1A-6
Reactor Area 1A-5
Reactor Area Common 1AB
Reactor Area 1B-5
Reactor Area 1B-6
Loads
Loads
Loads
Loads
Loads
Battery Charger 1AA
Battery A
Battery Charger 1A
Battery Charger 1AB
Battery Charger 1B
Battery B
Battery Charger 1BB
125 V DC Level
1A
1AB
1B
Loads
Loads
Inverters
1A
1C
1B
1D

Figure 3.1 Onsite Electric Power System Schematic

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failure level for ceramic insulators (which causes seismic LOSP) failure of the ceramic insulators is the dominant cause of lack of availability of the main feedwater system.

Auxiliary Feedwater System

This consists of a water source comprised of the two condensate storage tanks (CSTs) and two motor driven pumps and one turbine driven pump, as shown in Figure 3.2. The turbine pump requires only DC power from DC bus AB for control. The steam supply to the turbine pump is shown in Figure 3.3.

No single failures were identified for this system. From a seismic viewpoint, the important failure modes are as follows:

a) Loss of both CSTs
b) Loss of offsite power plus loss of both battery banks. Since diesel generators cannot start without DC power to flash their fields, no emergency AC or DC power is available. Hence seismically induced station blackout results, and all three AFW pumps are unavailable.
c) Loss of diesel generator 1 and loss of battery B given LOSP. Loss of DG1 fails the AFWS motor pump 1A while loss of the B battery bank fails DG2 (and hence motor pump 1B) and additionally DC bus AB is not energized, and thus no control power to the turbine pump is available. Thus all three AFWS pumps are unavailable.
d) Common mode failure of both diesel generators (given LOSP) and loss of either the AFW turbine pump or its supply valve AFW803. The latter fails the turbine pump train, while failure of both diesels fails both motor driven AFW pumps.

High Pressure Injection System

A schematic for this system is shown in Figure 3.4. Water is pumped from the refueling water storage tank (RWST) to the primary system by three pumps (1A, 1B, 1C), each requiring AC motive power and DC SIS actuation power. Pump 1A is fed by electrical train A, while pumps 1B and 1C are fed by train B. All pumps require cooling from the component cooling water system (CCW), provided as shown in Figure 3.5. Note that pump CCW-P1C is normally valved out, and CCW-P1A cools HPI pump 1A, while CCW-P1B cools both HPI pumps 1B and 1C. (The same is true for the LPSI and CSI pumps cooling.) Finally, the CCW heat exchangers HTX-1A and -1B are cooled respectively by the Intake Cooling Water (ICW) trains A and B respectively, as shown on Figure 3.6.

Important seismic failure modes for this system are as follows:

a) Failure of RWST
b) Failure of CCWS accumulator tank or heat exchangers
c) Common-mode failure of SIS sensors

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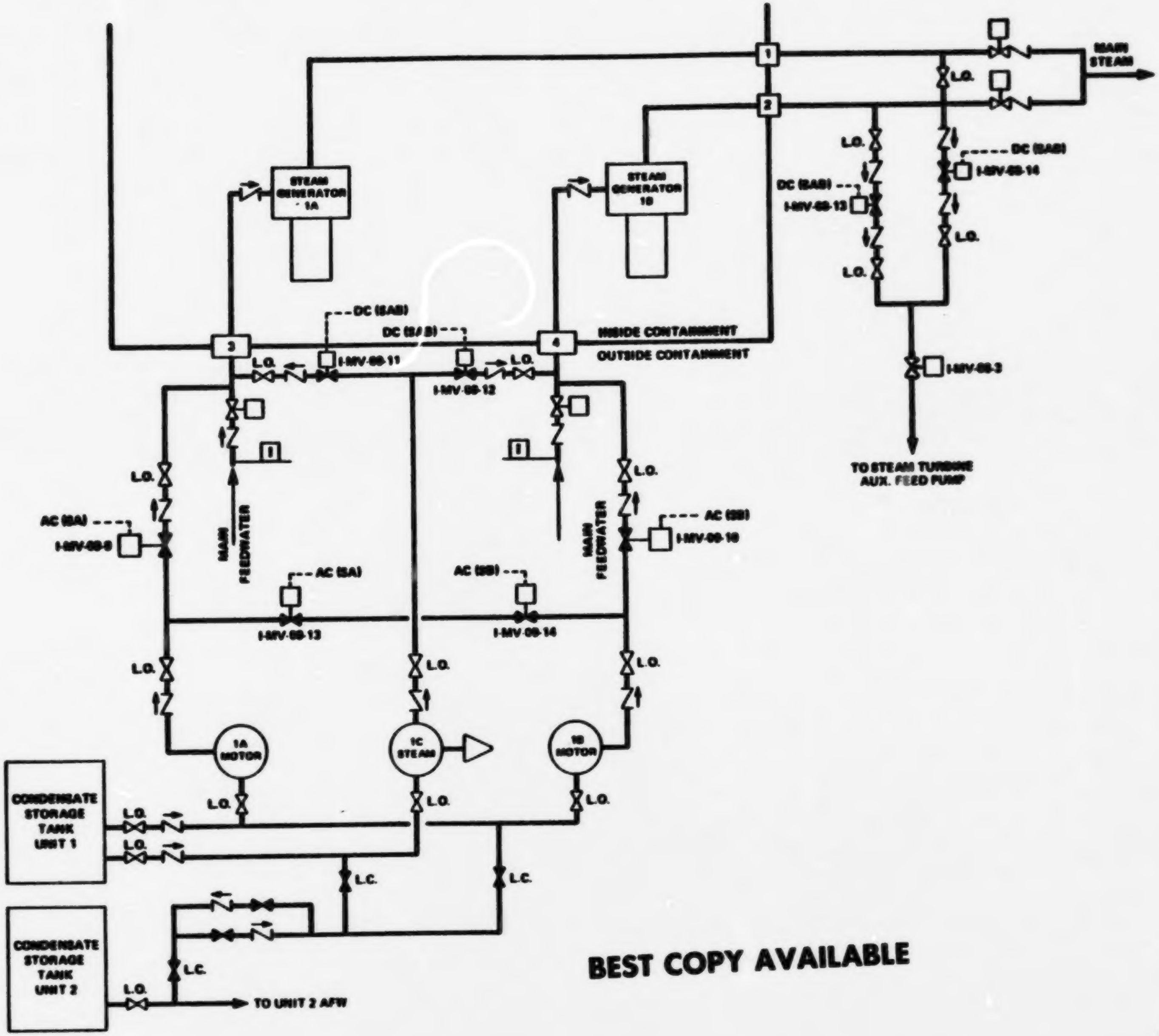

MAIN STEAM
L.O.
1
2
STEAM GENERATOR 1A
STEAM GENERATOR 1B
L.O.
DC (SAB)
I-MV-08-14
DC (SAB)
I-MV-08-13
L.O.
L.O.
I-MV-08-3
TO STEAM TURBINE AUX. FEED PUMP
DC (SAB)
DC (SA/B)
3
4
INSIDE CONTAINMENT
OUTSIDE CONTAINMENT
L.O.
I-MV-09-11
I-MV-09-12
L.O.
L.O.
AC (SA)
I-MV-09-9
MAIN FEEDWATER
MAIN FEEDWATER
L.O.
AC (SB)
I-MV-09-10
AC (SA)
AC (SB)
I-MV-09-13
I-MV-09-14
L.O.
L.O.
L.O.
1A MOTOR
1C STEAM
1B MOTOR
L.O.
L.O.
L.O.
CONDENSATE STORAGE TANK UNIT 1
L.O.
L.C.
L.C.
CONDENSATE STORAGE TANK UNIT 2
L.C.
L.O.
TO UNIT 2 AFW

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Figure 3.2 Auxiliary Feedwater System Schematic

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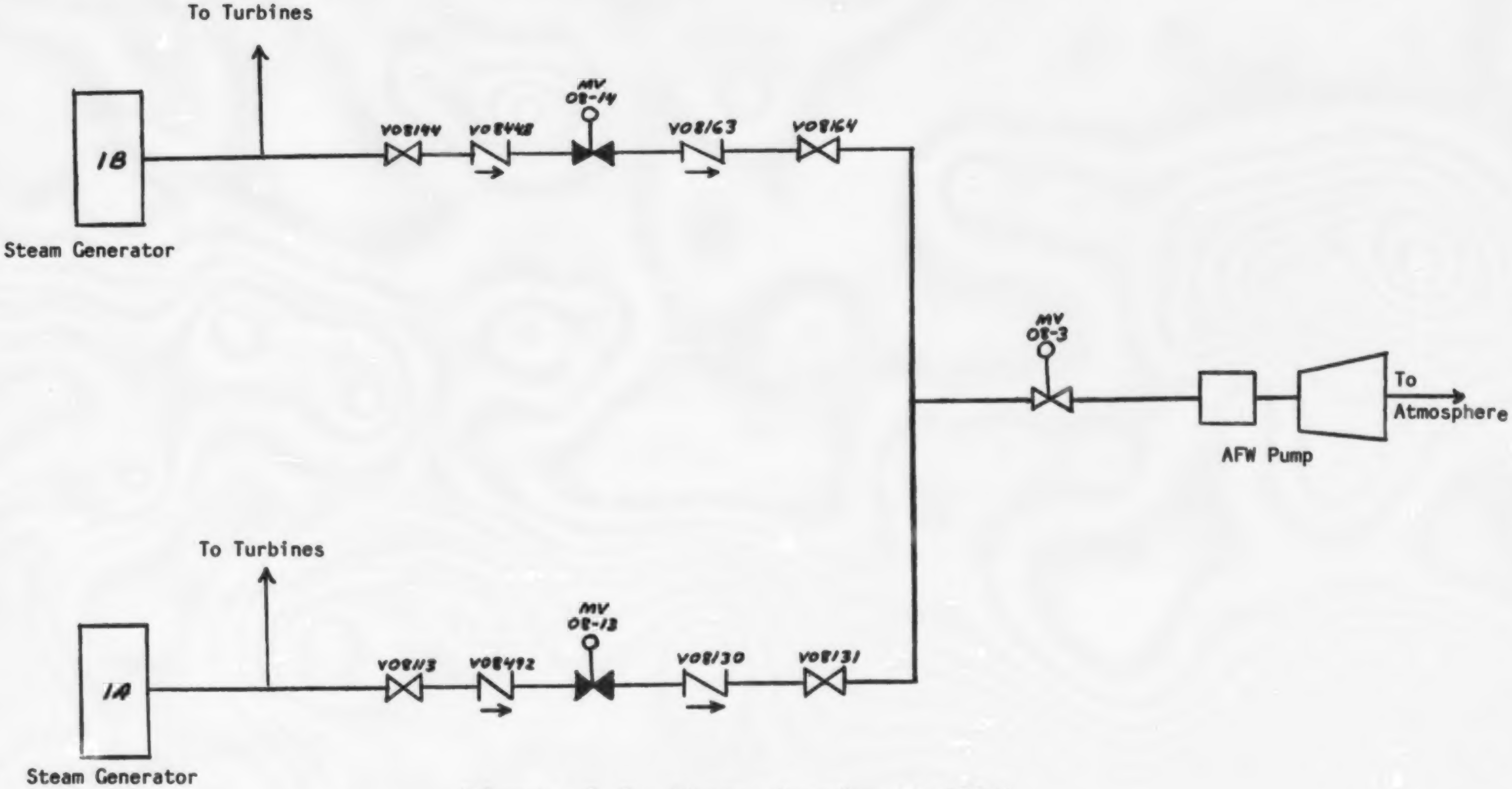
To Turbines
1B
Steam Generator
V08144
V08448
MV
08-14
V08163
V08164
MV
08-3
To
Atmosphere
AFW Pump
To Turbines
1A
Steam Generator
V08113
V08492
MV
08-13
V08130
V08131

Figure 3.3 Steam Supply to AFWS

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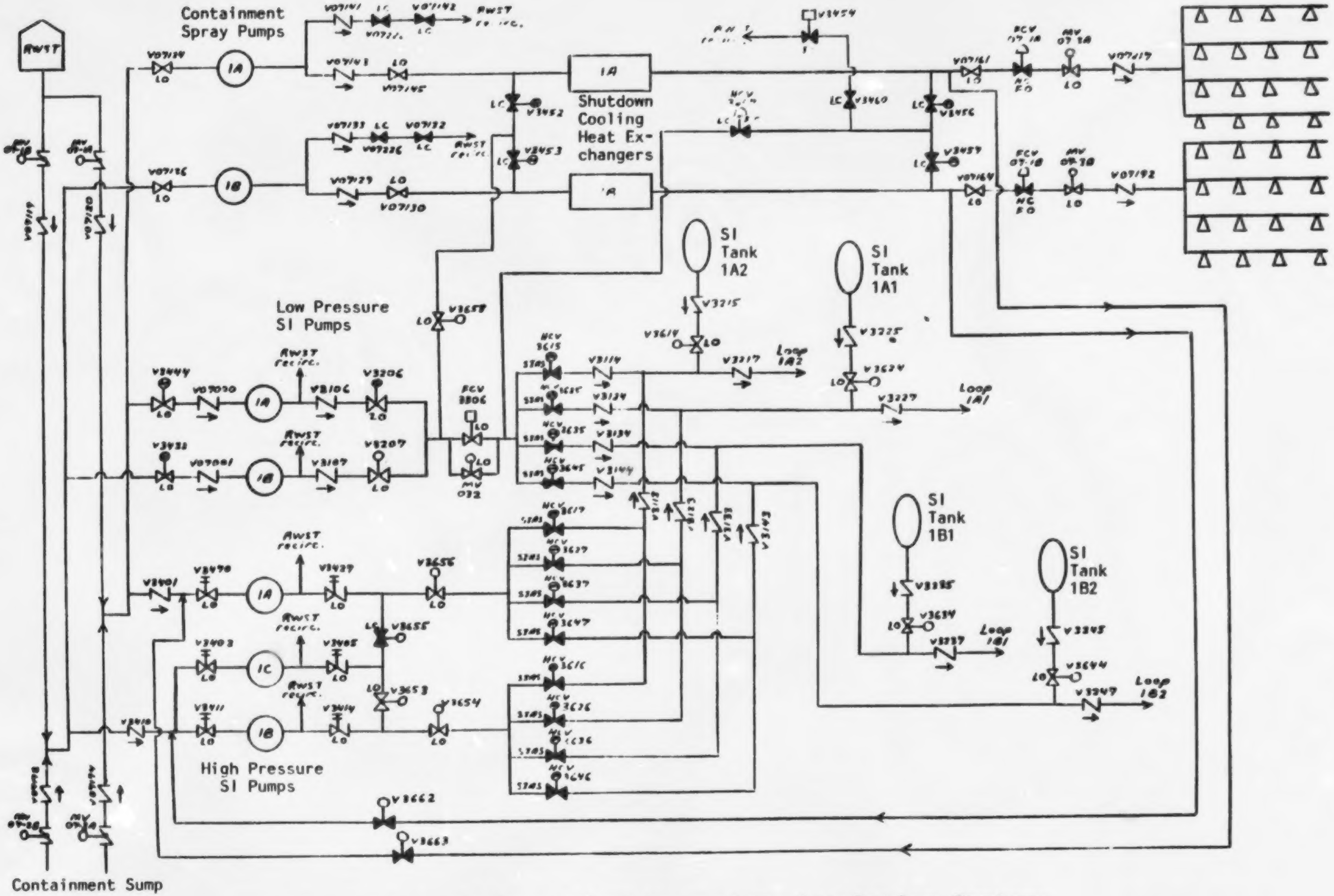
RWST
Containment Spray Pumps
1A
1B
RWST Recirc.
V3452
V3453
1A
Shutdown Cooling Heat Exchangers
V3454
V3460
V3456
V3457
LC
LO
NC
FO
SI Tank 1A2
SI Tank 1A1
V3215
V3225
Loop 1A2
Loop 1A1
Low Pressure SI Pumps
1A
1B
V3444
V3432
V3106
V3107
V3206
V3207
V3658
FCV 3306
MV 032
SIAS
SI Tank 1B1
SI Tank 1B2
V3235
V3245
Loop 1B1
Loop 1B2
V3401
V3470
V3427
V3656
V3655
V3653
V3654
V3403
V3405
V3411
V3414
V3410
1A
1C
1B
High Pressure SI Pumps
V3662
V3663
Containment Sump

Figure 3.4 Safety Injection and Recirculation Systems

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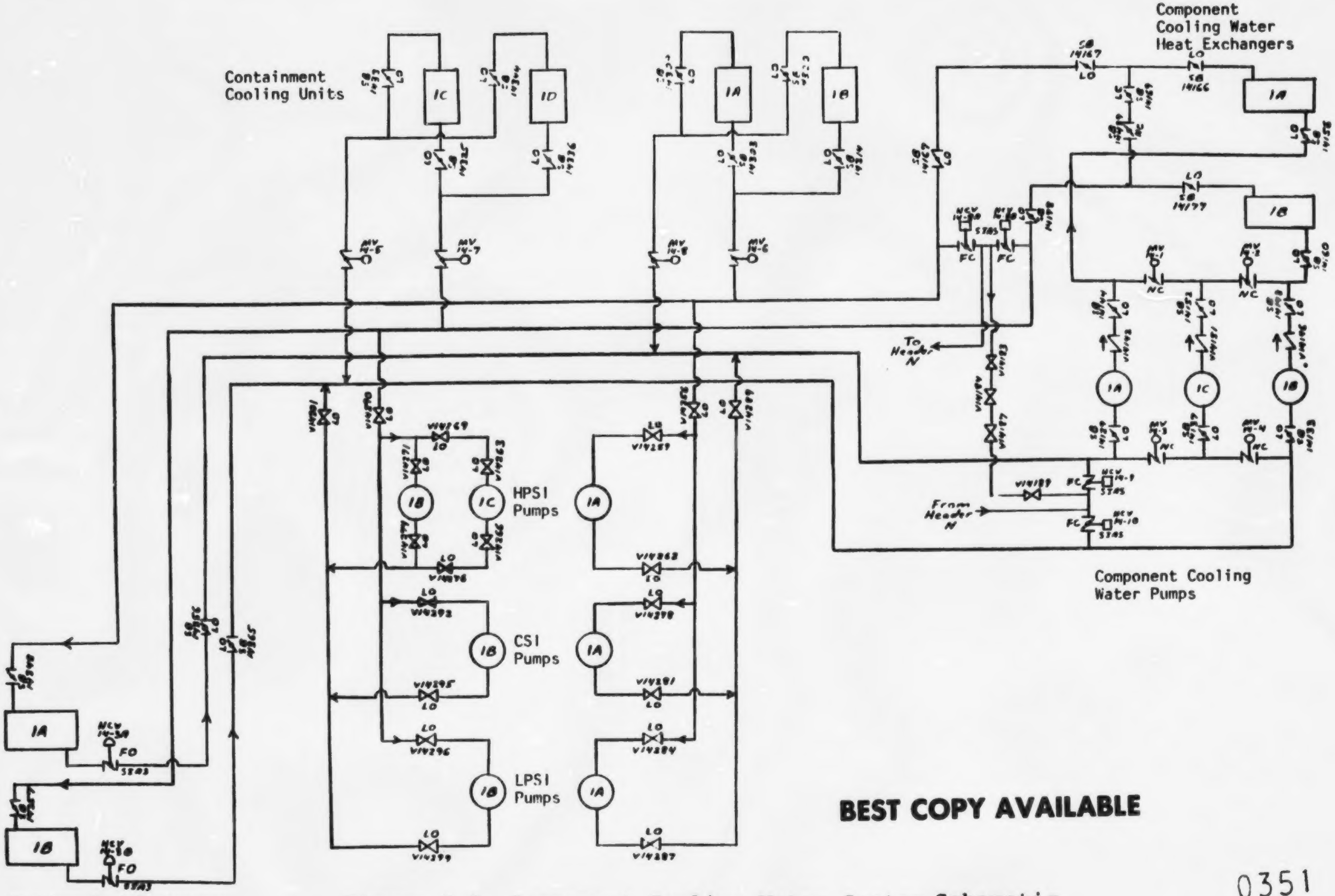
Containment Cooling Units
1C
1D
1A
1B
Component Cooling Water Heat Exchangers
1A
1B
SB 14167
SB 14166
SB 14177
LO
LC
NC
LT
MV 14-5
MV 14-7
MV 14-8
MV 14-6
MV 14-1
MV 14-2
MV 14-3
MV 14-4
HCV 14-8
HCV 14-9
HCV 14-10
STAS
FC
To Header N
From Header N
V14189
1A
1C
1B
Component Cooling Water Pumps
V14269
V14276
V14267
V14262
1B
1C
1A
HPSI Pumps
V14292
V14295
V14278
V14281
1B
1A
CSI Pumps
V14296
V14299
V14284
V14287
1B
1A
LPSI Pumps
1A
1B
HCV 14-3A
HCV 14-3B
FO
STAS
Shutdown Heat Exchangers

Figure 3.5 Component Cooling Water System Schematic

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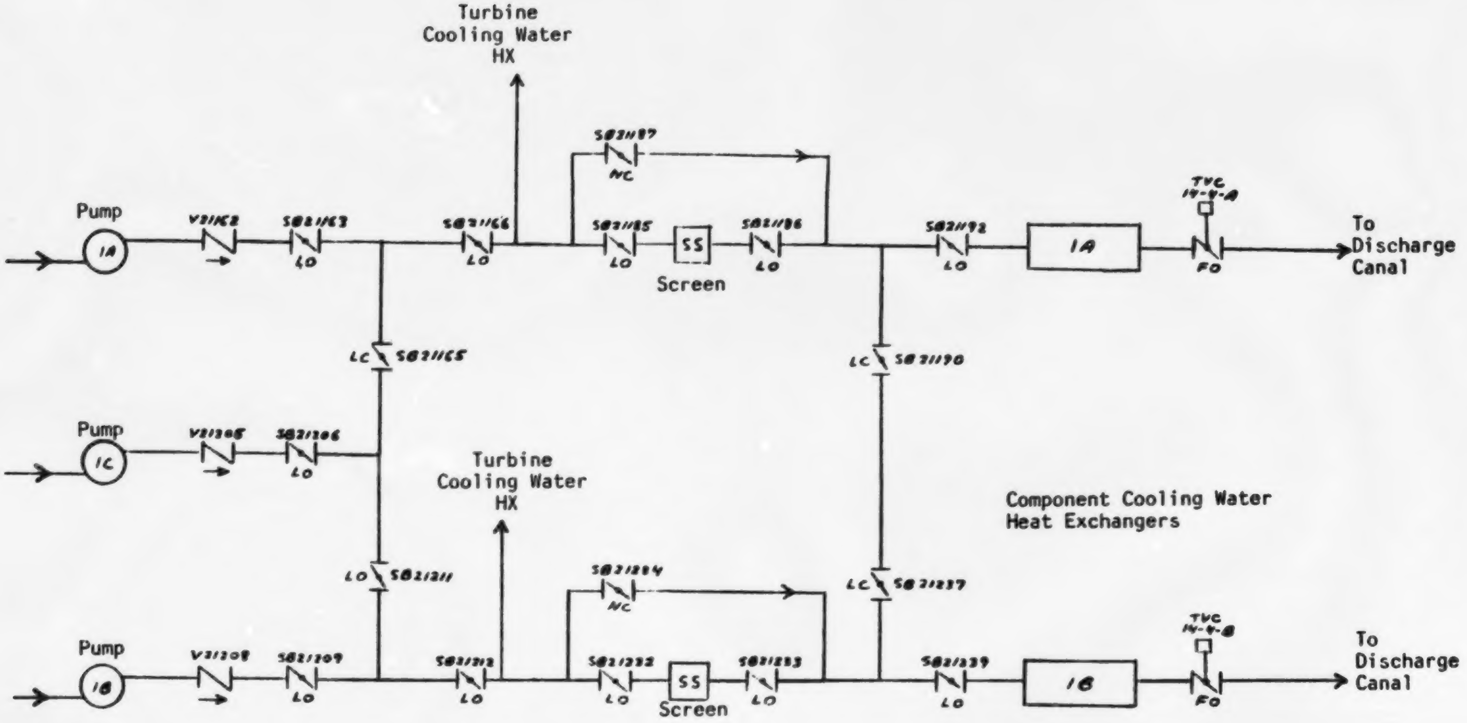
Turbine Cooling Water HX
SB21/87
NC
Pump
1A
V21/62
SB21/63
LO
SB21/66
LO
SB21/85
LO
SS
Screen
SB21/86
LO
SB21/92
LO
1A
TVC 14-4-A
FO
To Discharge Canal
LC SB21/65
LC SB21/90
Pump
1C
V21/305
SB21/306
LO
Turbine Cooling Water HX
Component Cooling Water Heat Exchangers
LO SB21/211
SB21/234
NC
LC SB21/237
Pump
1B
V21/208
SB21/209
LO
SB21/212
LO
SB21/232
LO
SS
Screen
SB21/233
LO
SB21/239
LO
1B
TVC 14-4-B
FO
To Discharge Canal

Figure 3.6 Intake Cooling Water System

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d) Loss of both ICW pumps or trains, or loss of both (on-line) CCW pumps or trains or any train A-train B combination thereof.
e) Loss of one train of either CCW or ICW and the opposite train of electrical AC power.
f) Loss of both shutdown (RHR) heat exchangers or both CCW heat exchangers.
g) Loss of both batteries in conjunction with loss of off-site power

High Pressure Recirculation System

This system makes use of the same high pressure pumps, except suction is now taken from the two sumps through a common header, as shown in Figure 3.4. The failure modes are essentially the same as for the HPI system.

Low Pressure Injection

This system, shown in Figure 3.4, is essentially the same as the High Pressure Injection System, having two AC-driven RHR pumps and taking water from the RWST. All the failure modes identified for the High Pressure Injection System apply here also.

Low Pressure Recirculation System

This system is essentially the same as the LPI system, except that suction is taken from the two sumps, as shown in Figure 3.4. All previously identified failure modes apply here also.

Feed and Bleed and the Power-Operated Relief Valves

Feed and bleed (FNB) is considered a viable option at St. Lucie. In this operation, the HPI pumps are used to pump water into the primary coolant system, and steam (or two-phase flow) is released through the pilot-operated relief valves (PORVs). All the failure modes identified for the HPI system apply here also. For St. Lucie, it is assumed that both PORVs must open on demand, and that both block valves are closed during normal operation. Thus failure of a single block valve or a single PORV is assumed to thwart the bleed and feed operation. Similarly, the loss of either train of DC power is assumed to fail FNB.

Other Systems

The Intake Cooling Water System (the service water system) schematic is shown in Figure 3.6 for completeness. Finally, schematics of the containment spray and recirculation systems are also shown in Figure 3.4.

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4.0 SEISMICITY AND HAZARD CURVES

The St. Lucie nuclear power plant is located in a region which is considered seismically benign. In fact, in terms of seismic damage zones estimated from historical damaging earthquakes, St. Lucie is in the lowest United States risk zone, as seen on Figure 4.1 (Reference 9). According to the St. Lucie FSAR (Reference 8), there have been no more than 7 shocks in the past 200 to 250 years with epicenters located in Florida. Two of these had epicentral intensities of no more than VI (Modified Mercalli). Other shocks include those in Cuba (approximately 250 miles to the south) and the largest shock in the general region was the MM X Charleston, South Carolina shock of 1886.

The Safe Shutdown Earthquake at St. Lucie was taken as 0.10g horizontal, defined at the -34 foot foundation elevation. The Operating Basis Earthquake was taken as 0.05g horizontal. In both cases, vertical accelerations were taken as 2/3 of the horizontal accelerations. The value of 0.10g for the SSE resulted from the Appendix A 10 CFR Part 100 requirement of a minimum SSE of 0.10g for any site. Accelerations at the St. Lucie site from actual recorded historical earthquakes were estimated in the range of 0.03g to 0.07g.

In performing seismic risk assessments for the plants being studied as part of TAP A-45, we have adopted the procedure of setting the SSE peak ground acceleration to an annual probability of exceedance of 2.5×10^{-4} per year. This is only appropriate, however, if the SSE is determined from the recorded site seismicity and according to the guidelines of Appendix A to 10 CFR Part 100. Thus, before using the value of 0.10g to generate a hazard curve at St. Lucie, we must evaluate the level of ground acceleration which would be required at the St. Lucie site without the 0.10g minimum SSE requirement.

Given that the largest shocks in the last 200 years on the Florida peninsula were of magnitude MM VI, we can use the Trifunac and Brady (Reference 10) relation to relate ground acceleration to intensity, i.e.,

$$\log A_h = -0.014 + 0.3\, I_{mm}$$

where

A_h = peak horizontal ground acceleration (cm/sec^2)

I_{mm} = modified Mercalli intensity

which gives $A_h = 0.062g$ for I_{mm} = VI. Another commonly accepted relation is that of Murphy and O'Brien, Reference 11, which yields $A_h = 0.051g$. These are in close accord with the value of 0.05g derived in the St. Lucie FSAR for an intensity

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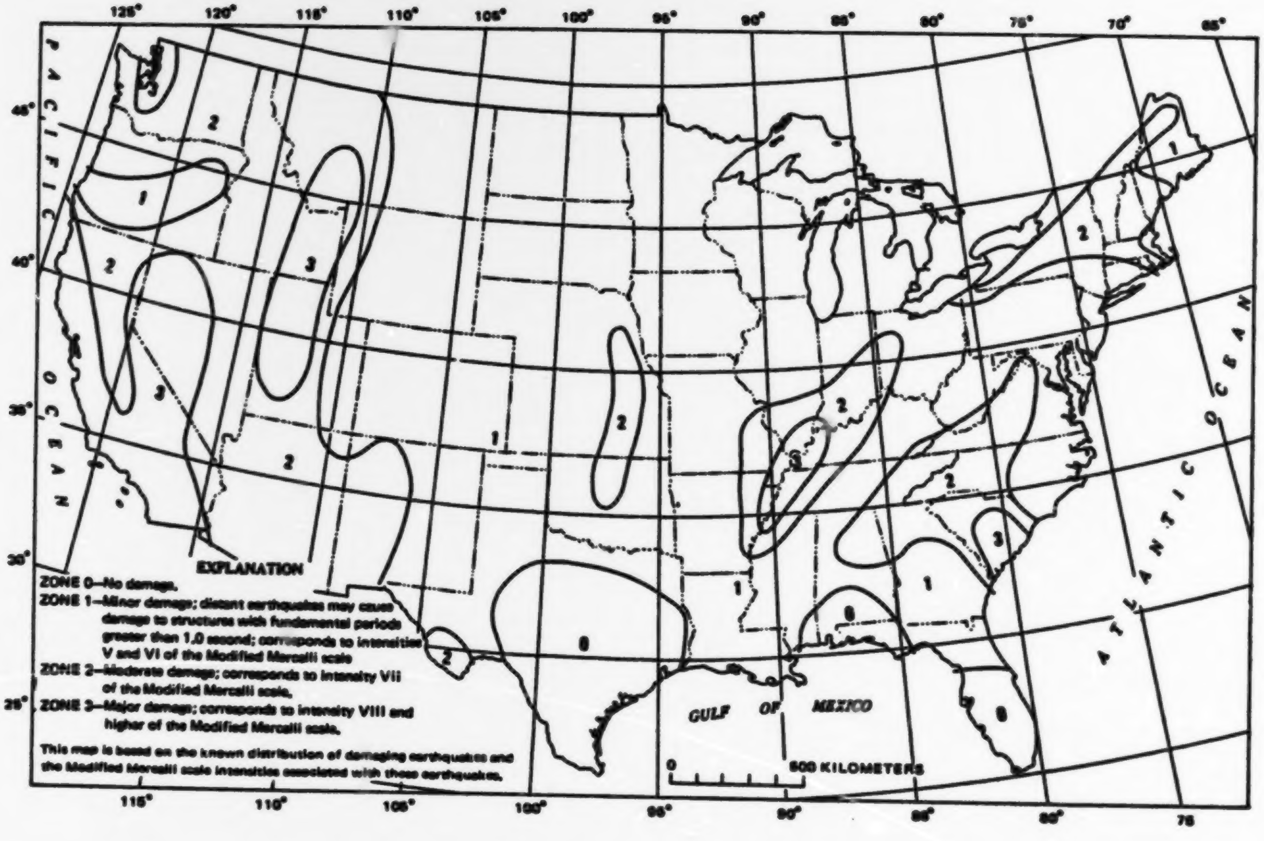
125°
120°
115°
110°
105°
100°
95°
90°
85°
80°
75°
70°
65°
45°
40°
35°
30°
25°
PACIFIC OCEAN
ATLANTIC OCEAN
GULF OF MEXICO
0
500 KILOMETERS
EXPLANATION
ZONE 0—No damage.
ZONE 1—Minor damage; distant earthquakes may cause damage to structures with fundamental periods greater than 1.0 second; corresponds to intensities V and VI of the Modified Mercalli scale
ZONE 2—Moderate damage; corresponds to intensity VII of the Modified Mercalli scale.
ZONE 3—Major damage; corresponds to intensity VIII and higher of the Modified Mercalli scale.
This map is based on the known distribution of damaging earthquakes and the Modified Mercalli scale intensities associated with those earthquakes.
115°
110°
105°
100°
95°
90°
85°
80°
75

Figure 4.1 United States Seismic Risk Zones (Reference 9)

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MM VI earthquake using an intensity-acceleration relation due to Hershberger. Thus, peak ground accelerations due to the few historical earthquakes which have occurred on the Florida penninsula in the range of 0.05-0.07g appear reasonable. However, accelerations at the St. Lucie site due to historical earthquakes in surrounding tectonic zones must also be considered.

The next nearest major historical earthquake was the Charleston, South Carolina earthquake of 1886. Assuming an intensity X MM earthquake in Zone 65 of Figure 4.2 (from Reference 10) and the closest distance from Zone 65 to the St. Lucie site of 318 km, in conjunction with the Gupta-Nuttli attenuation relation (Reference 11)

$$I = I_0 + 3.7 - 0.001R - 2.7 \log R \ (R > 10 \text{ km})$$

gives an intensity of 6.6MM at the site. Converting this to peak ground acceleration using the Trifunac & Brady equation cited above gives a value of horizontal peak ground acceleration of 0.09g. This is sufficiently close to the St. Lucie SSE value of 0.10g to conclude that 0.10g is an appropriate SSE value to use in developing a hazard curve for St. Lucie for use in TAP A-45.

Thus, the hazard curve used in this seismic risk evaluation of the St. Lucie nuclear power plant is based on prescribing a hazard curve with the 0.10g SSE fixed at an annual probability of exceedance of 2.5E-4 per year as described in Reference 2. The slope of the curve is taken from hazard curves developed for nearby sites.

Figure 4.3 shows nearby sites at which hazard investigations have been made in the Eastern U.S. Seismic Hazard Characterization program (Reference 12). The Vogtle and River Bend sites are the closest, and their hazard curves are shown in Figure 4.4.

It can be seen that the slopes of these two curves are very nearly the same. The hazard curve at St. Lucie was thus constructed by shifting the (nominal) River Bend hazard curve vertically downward so that the SSE of 0.10g is fixed at 2.5E-4 per year. This shifted curve is also shown in Figure 4.4, and will be referred to as the nominal rock outcrop hazard curve for the St. Lucie site.

An important feature at any soil site, however, is to consider potential corrections to the nominal rock outcrop hazard curve due to the local site conditions. The St. Lucie site is a deep soil site with a soil column consisting of 78 feet of engineering fill over 540 feet of natural sands and clay-sand deposits, all of which overlay a limestone bedrock. As discussed in detail in Chapter 5.0, the response of this soil column at St. Lucie was computed at three different rock outcrop

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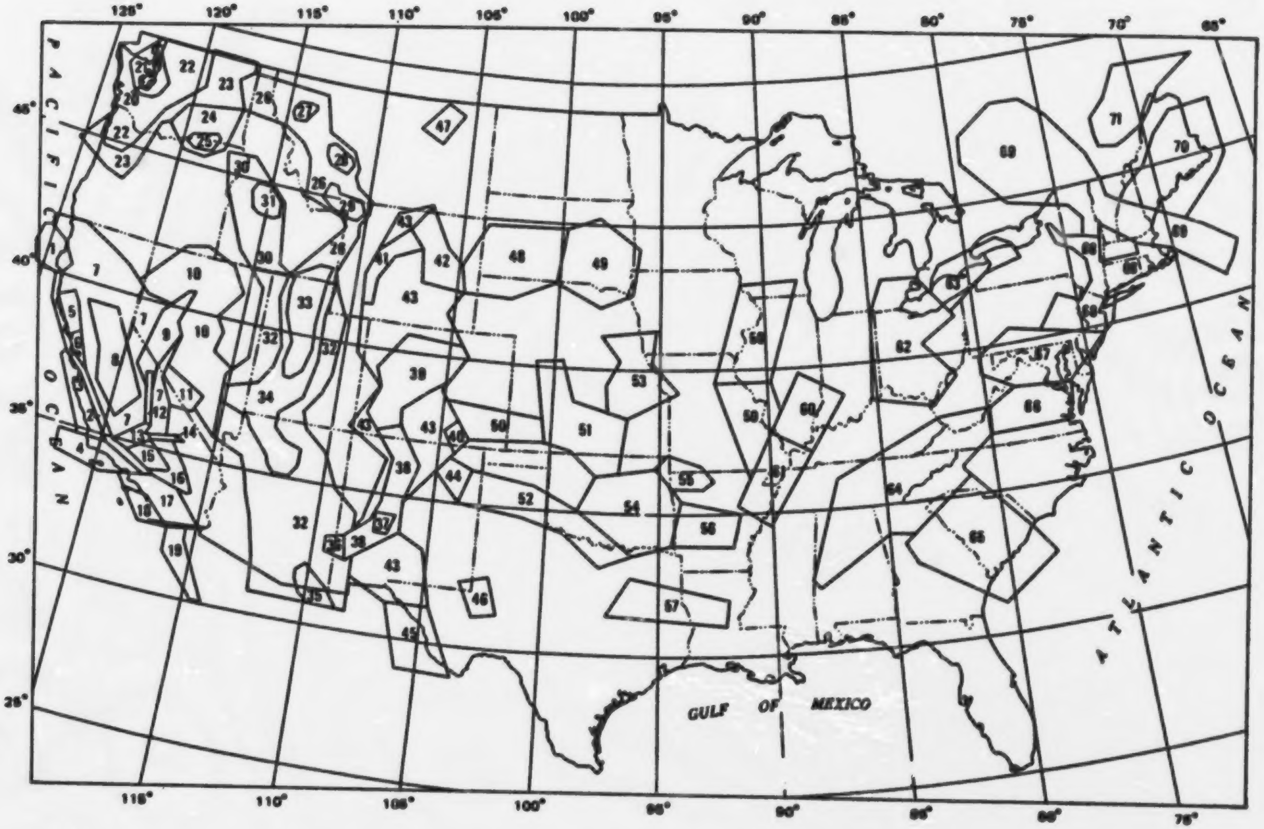
125°
120°
115°
110°
105°
100°
95°
90°
85°
80°
75°
70°
65°
45°
40°
35°
30°
25°
115°
110°
105°
100°
95°
90°
85°
80°
75°
PACIFIC OCEAN
ATLANTIC OCEAN
GULF OF MEXICO

Figure 4.2 Seismic Source Zones (Algermissen and Perkins, 1976)

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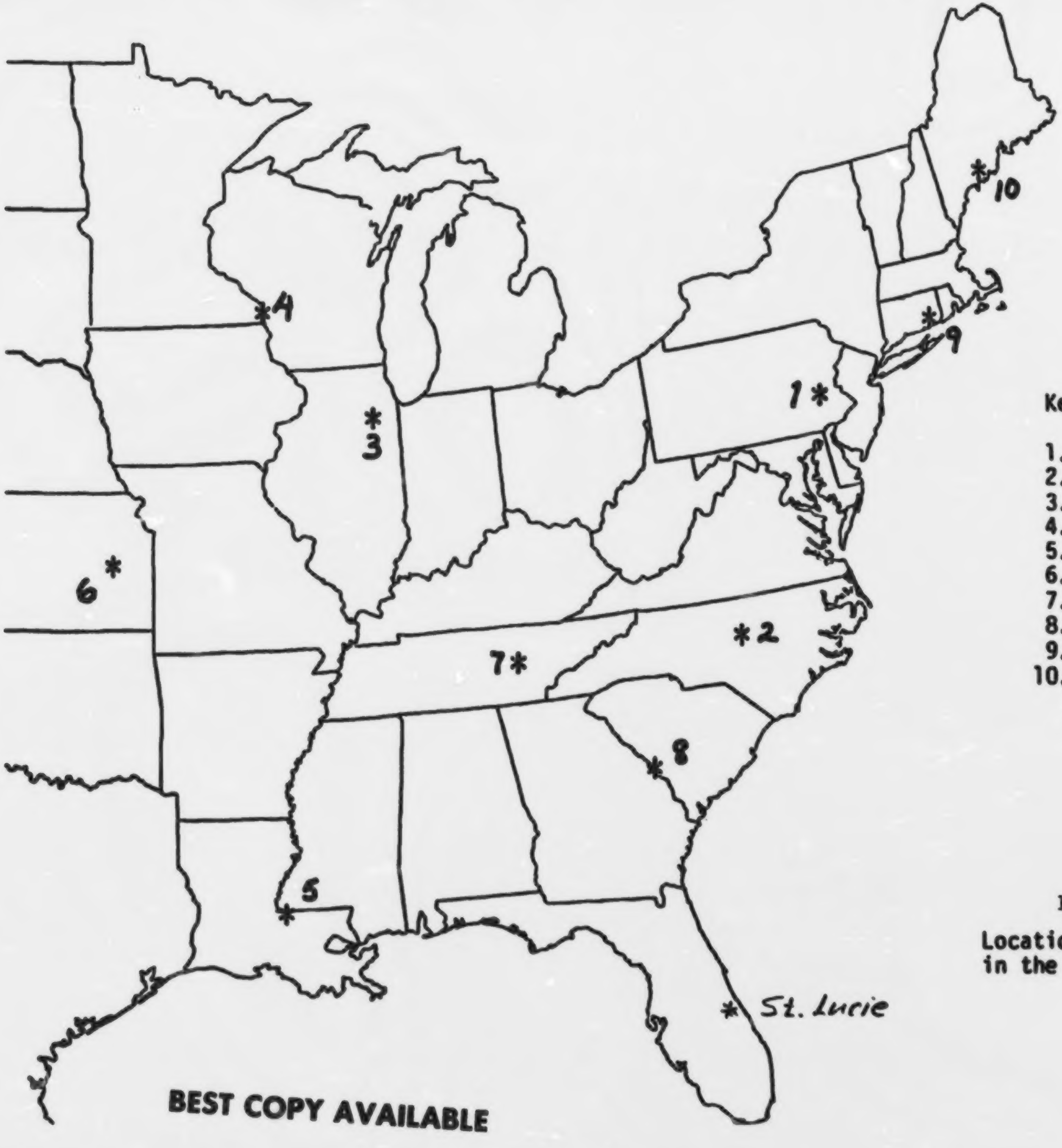

Key to Site Index Numbers

1. Limerick
2. Shearon Harris
3. Braidwood
4. La Crosse
5. River Bend
6. Wolf Creek
7. Watts Bar
8. Vogtle
9. Millstone
10. Maine Yankee

Figure 4.3

Location of the Sample Sites in the EUS

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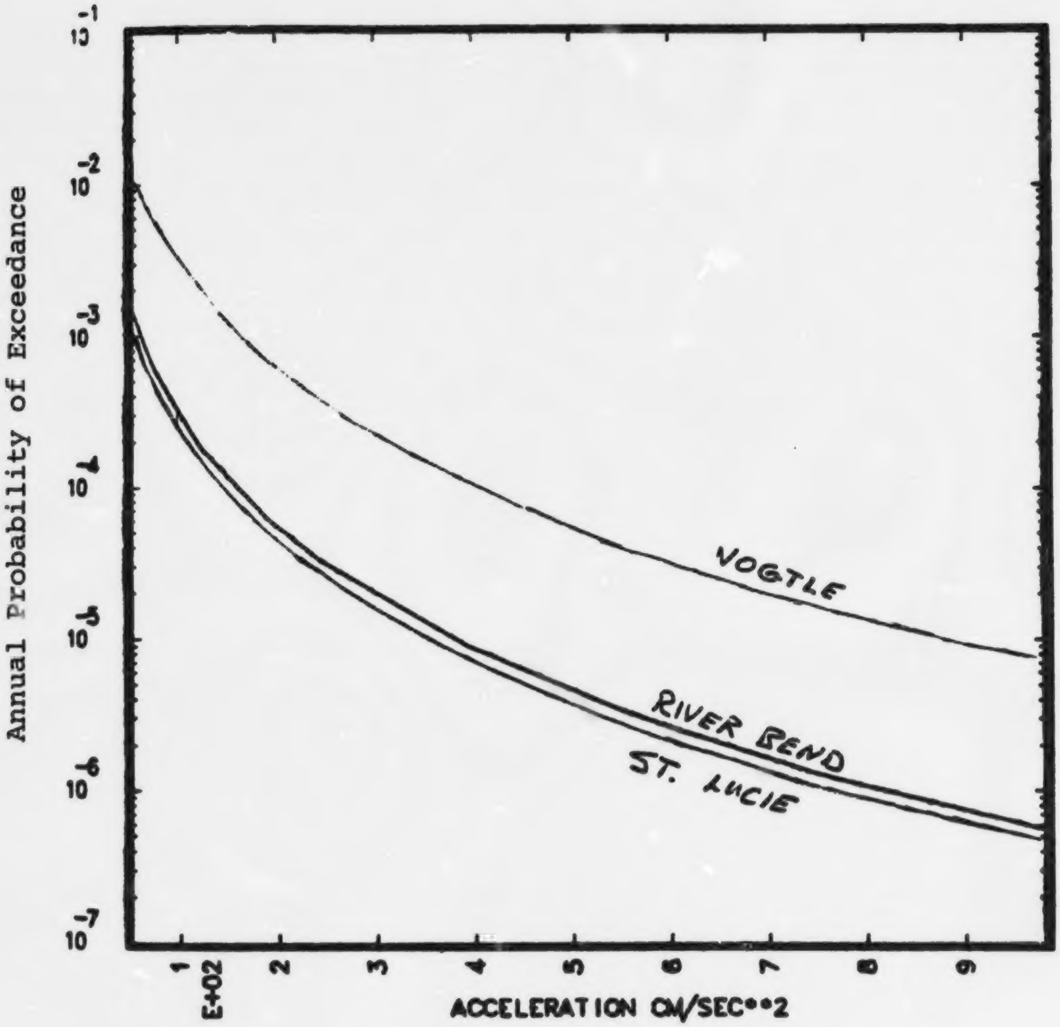

Figure 4.4 Hazard Curves for St. Lucie and Nearby Sites

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earthquake levels (0.10g, 0.20g, and 0.30g), using the SHAKE code (Reference 5) to compute the response of the soil column. From these calculations, the ratio of the rock outcrop to surface peak ground accelerations was computed as shown in Table 4.1. This ratio varied from 1.1 at the lowest level to 0.87 at the highest level. Thus it is seen that very little amplification is expected at the lower earthquake levels, and some slight de-amplification is expected at the higher earthquake levels. Thus, for practical purposes, no local site correction is needed for the St. Lucie hazard curve.

The St. Lucie hazard curve, then, obtained by a downward vertical shift of the nominal River Bend curve, is shown in Figure 4.4. Numerical values of the St. Lucie hazard curve are listed on Table 4.2. From these values, the probabilities of having an earthquake in the 1-2 SSE, 2-3 SSE, 3-4 SSE, and 4-5 SSE ranges were obtained, as given in Table 4.3.

Table 4.1 St. Lucie PGA Amplification Factors due to Local Site Effects

Rock Outcrop	Surface	Outcrop to Surface PGA Ratio
0.10g	0.11g	1.10
0.20g	0.19g	0.95
0.30g	0.26g	0.87

Table 4.2 St. Lucie Hazard Curve Coordinates

PGA	Annual Probability of Exceedance
0.10	2.5E-4
0.20	5.0E-5
0.30	1.9E-5
0.40	8.0E-6
0.50	4.0E-6

Table 4.3 Probabilities Per Year of Hazard Curve Levels at St. Lucie

Earthquake Level	Surface PGA Interval	Probability Per Year
1	0.10 - 0.20	2.0E-4
2	0.20 - 0.30	3.1E-5
3	0.30 - 0.40	1.1E-5
4	0.40 - 0.50	4.0E-6

0361

5.0 RESPONSE CALCULATIONS

5.1 Introduction

As described previously, seismic probabilistic risk assessments (PRA's) can be considered in a series of steps: seismic hazard characterization, seismic response of structures and components, structure and component failure descriptions, plant logic models, and probabilistic failure calculations. Section 5 deals with the frequency characteristics of the free-field ground motion (an element of the seismic hazard characterization) and the seismic response of structures and components.

In a seismic PRA of decay heat removal (DHR) systems, only the components affecting the operation of the systems and those structures housing or supporting these components need to be analyzed. Plant logic models identify the components. Plant general arrangement and mechanical drawings are then used to locate the components and identify the relevant supporting structures. For the St. Lucie Nuclear Power Plant the specific components of concern to the DHR systems and the component locations had not been identified at the time of this seismic response assessment. However, previous experience in the analysis of DHR systems at similar plants provided sufficient insight to identify the structures to be analyzed. For this study, seismic analyses of the reactor auxiliary building and diesel generator building were performed. Figure 5.1 illustrates the general plant layout and shows relative location of these structures.

Seismic PRA's require as input best-estimate structural response, variations of response, and correlation of response. Recall that the seismic PRA considers earthquakes over the entire range of the seismic hazard curve, hence, seismic responses must be determined over this range. Often, seismic response determined as part of the plant design process is available. However, these data reflect the conservatism associated with the seismic design analysis methodology and consider only low seismic excitation levels.

To determine structural response at the higher excitation levels required by a seismic PRA, either the design analyses must be extrapolated or reanalyses of the structures must be made. For this study, the latter choice was made. Mathematical models of the reactor auxiliary and diesel generator buildings developed by the utility for their design analyses were used in detailed probabilistic response analyses to determine the best-estimate seismic response of these structures.

The balance of this chapter will describe and summarize

- the site and earthquake characteristics that provide the starting point in the best-estimate soil-structure interaction analysis of the St. Lucie structures;
- the probabilistic response analysis of the reactor auxiliary building and the diesel generator building;
- the in-structure response that define the response of components associated with the decay heat removal systems.

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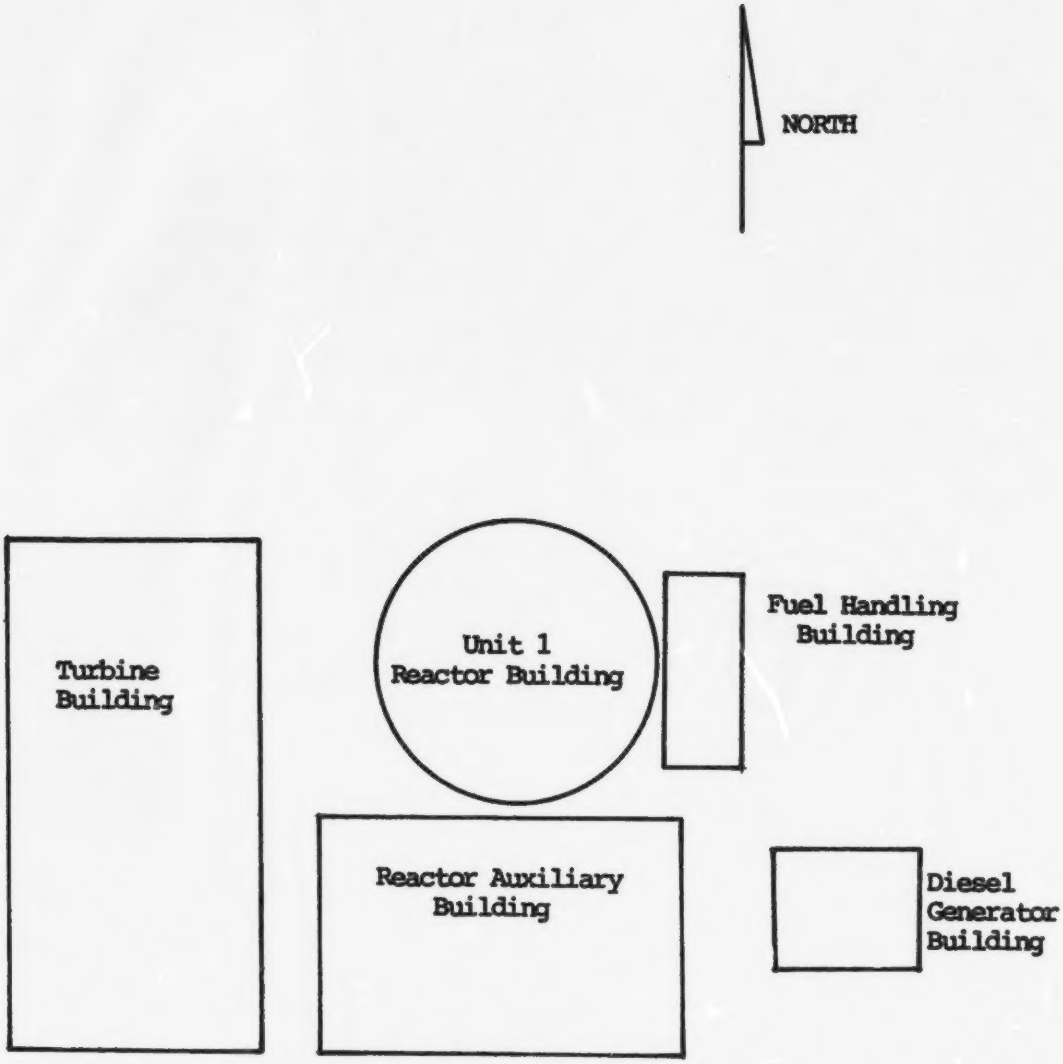
NORTH
Turbine Building
Unit 1 Reactor Building
Fuel Handling Building
Reactor Auxiliary Building
Diesel Generator Building

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Fig. 5.1 St. Lucie Nuclear Power Plant General Layout

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5.2 Site and Earthquake Characteristics

5.2.1 Site Description

The St. Lucie Nuclear Power Plant site in its natural state is an area where sand and clay-sand deposits overlie limestone. The layer of sand is an Anastasia formation extending from the surface at elevation 0'-0" to a depth of 145'. Below this is the clay-sand of the Hawthorne formation which extends to a depth of 600' where limestone is encountered. The top 60 ft. of soil was excavated and replaced with a Category I structural fill with grade at +18' elevation. Basemats of the structures of concern to this study were embedded and founded in the structural fill.

5.2.2 Soil Properties and Earthquake Definition

Two inter-related objectives of the initial portion of this investigation were to:

- define soil properties as a function of excitation level over the range of earthquakes to be considered, i.e., as defined by the seismic hazard curve.
- define the input motion for the determination of seismic responses of structure and components for this range of earthquakes

Figure 5.2 shows the low strain shear modulus and shear wave velocity profiles of the St. Lucie site as obtained from the FSAR [8]. A discontinuity in properties occurs between the structural fill and the Anastasia formation with a lesser discontinuity between the Anastasia and Hawthorne formations. Also, shown in Fig. 5.2 are the elevations of the Reactor Auxiliary Building (RAB) and Diesel Generator Building (DGB) foundations. These combinations of embedment depths and the soil profile complicates SSI modeling of the foundations. The strong layering effects potentially impact the seismic input due to the local site amplification.

Two cases were analyzed for the St. Lucie Reactor Auxiliary Building and the Diesel Generator Building, Neither for the two cases totally represent the St. Lucie site/foundations/SSI phenomena due to the limited availability of SSI parameters and the unique aspects of the St. Lucie site. As discussed, the site has a 78 ft. thick layer of engineered fill which has distinctly different properties than the underlying soils; hence, local site amplification is potentially important. Also, embedment is an important aspect of SSI for the RAB and DGB. SSI analyses performed here rely on previously calculated SSI parameters (impedances and scattering functions) due to budget constraints. The available SSI parameters for the St. Lucie structures embedment ratios are for uniform half-spaces; consequently SSI analyses for the two buildings were performed for half-space SSI parameters with

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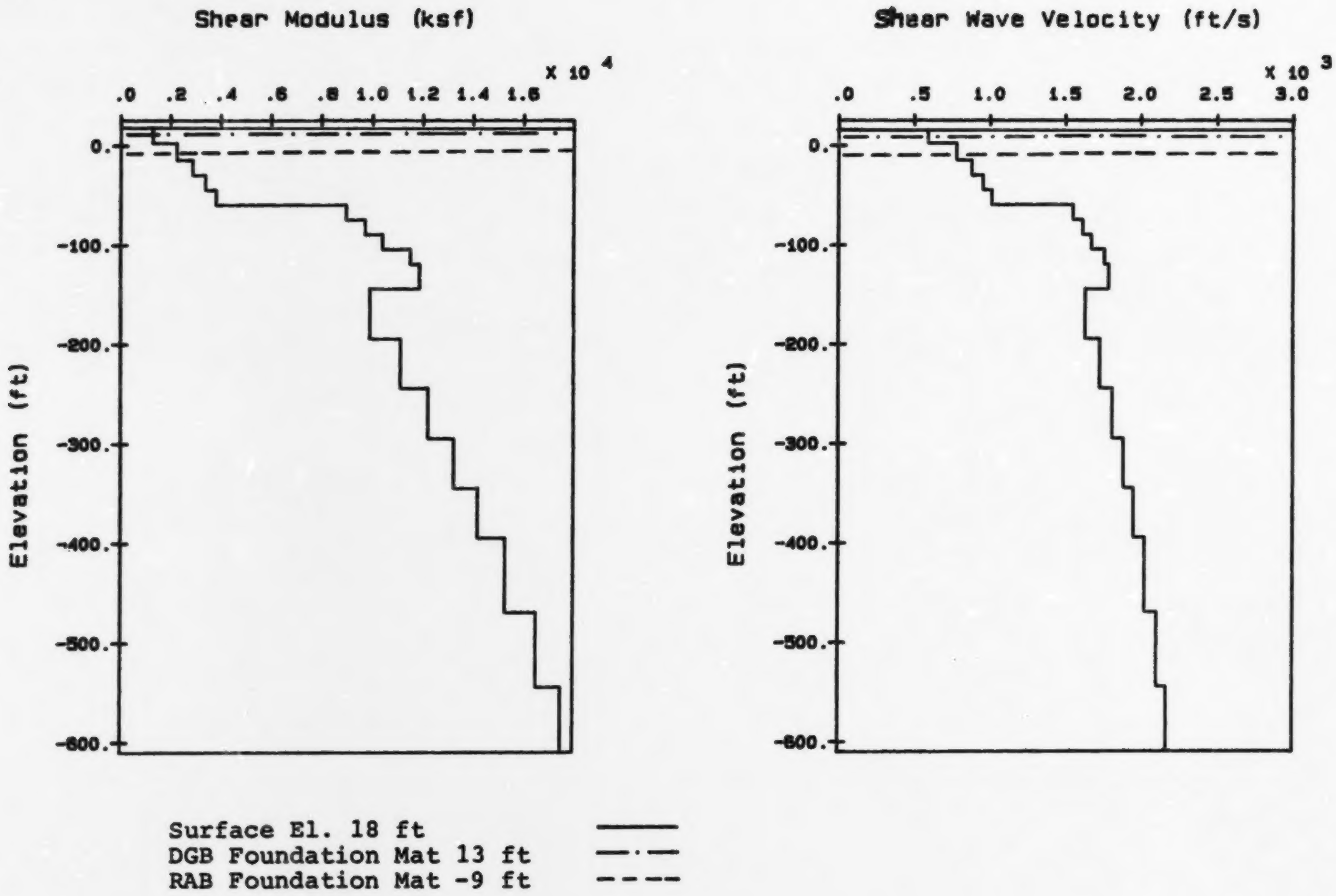

Fig. 5.2: Low Strain Shear Modulus and Shear Wave Velocity versus Depth

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average soil properties. In an attempt to assess the potential effects of local site amplification on in-structure response the two cases were run. Case 1 included local site amplification in the free-field ground motion. Case 2 assumed a free-field ground motion with broad-band frequency content.

Three earthquake levels were considered and defined by their peak ground acceleration in the horizontal direction -- 0.10g (SSE). 0.20g (2SSE), and 0.30g (3SSE). They are denoted acceleration ranges 1, 2 and 3 in subsequent discussions. These earthquake levels were treated explicitly -- input motions and seismic response for other levels defined by the hazard curve can be interpolated from the results.

In general, soil properties such as shear modulus and damping are a function of soil strain and consequently a function of earthquake excitation level. With higher excitation levels, soil shear modulus tends to decrease while soil damping tends to increase. Figures 5.3a and 5.3b show soil shear modulus and damping ratio vs. depth and their variation with acceleration range as determined by SHAKE [5] for Cases 1 and 2, respectively. Shear modulus degradation and damping variations with strain level assumed here were those of Ref.14. Cases 1 and 2 are now described in some detail.

Case 1

Local site amplification was taken into account by defining the seismic input on a hypothetical outcrop at elevation -60 ft. or 78 ft. beneath the top of grade. A single acceleration time history whose response spectrum matches median response spectra as presented in Ref.13 defined the motion on the outcrop. Figure 5.4 compares the median spectra from Ref. 4 with the response spectra generated from the artificial time history at 5% damping scaled to 1g. Three peak acceleration values were considered corresponding to the three acceleration ranges. A series of SHAKE [5] analyses were performed to define the motion on the soil free surface (top of grade, elevation +18 ft.) including variability of soil properties. Ten earthquakes were generated for each acceleration range with mean peak surface accelerations of .11g, .19g, and .26g; a slight deamplification of PGA's from outcrop is observed by the response spectra, lower frequencies show amplification. Figure 5.5 compares spectra generated from the outcrop time history with the mean of 10 free-field motions computed by SHAKE [2] for each acceleration range.

Case 2

The free-field ground motion frequency characteristics for Case 2 were defined by a median response spectra associated with deep alluvium sites and presented in Ref.14. A single time history analysis was performed for each acceleration level using nominal (best estimate) soil properties as determined by SHAKE [5]. Figure 5.6 compares the median deep alluvium spectra from Ref.14 with the response spectra generated from the artificial time history at 5% damping and scaled to 1g.

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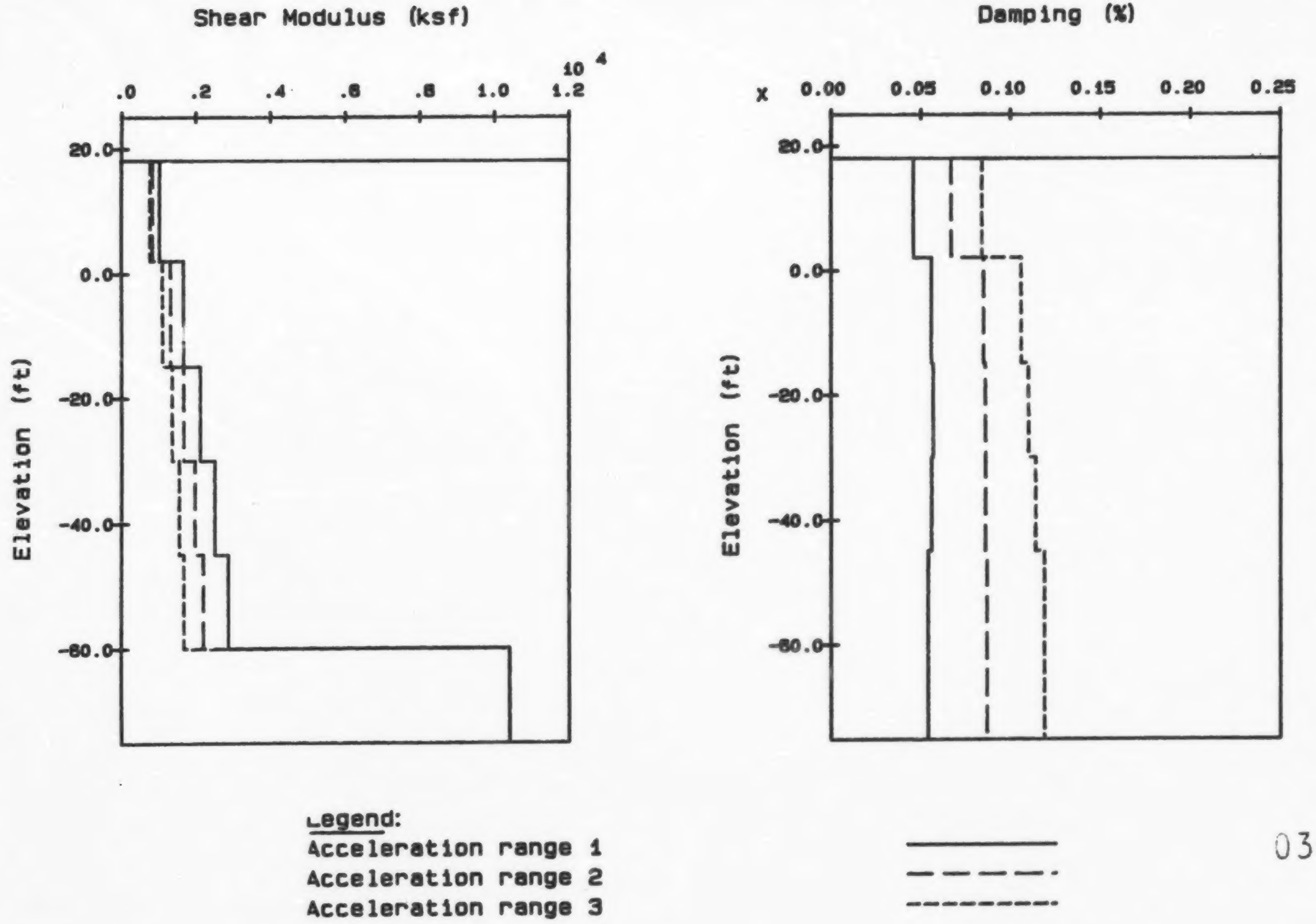

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Fig. 5.3a: Variation of Soil Shear Modulus and Damping Ratio With Depth - Case 1

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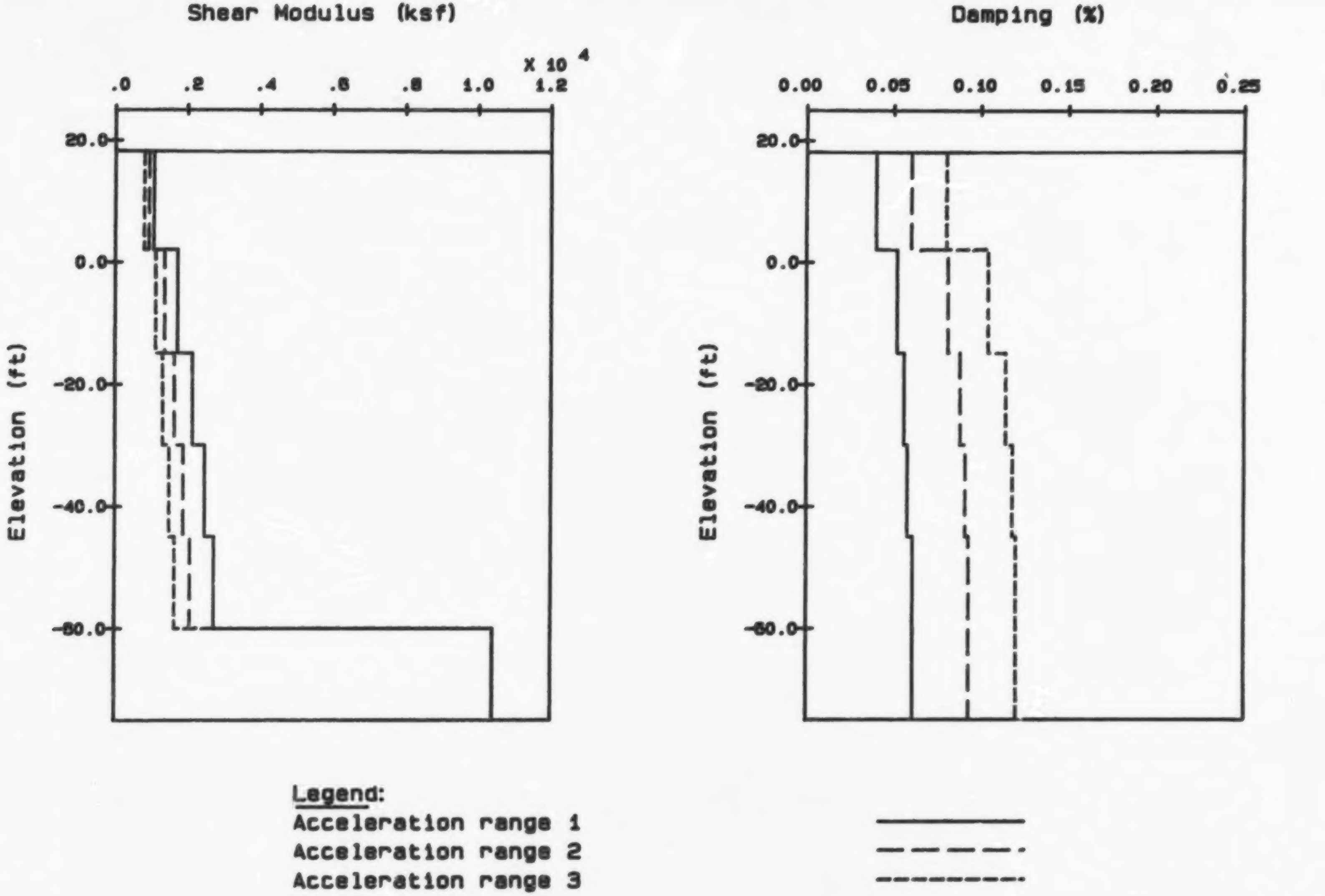

Fig. 5.3b: Variation of Soil Shear Modulus and Damping Ratio With Depth - Case 2

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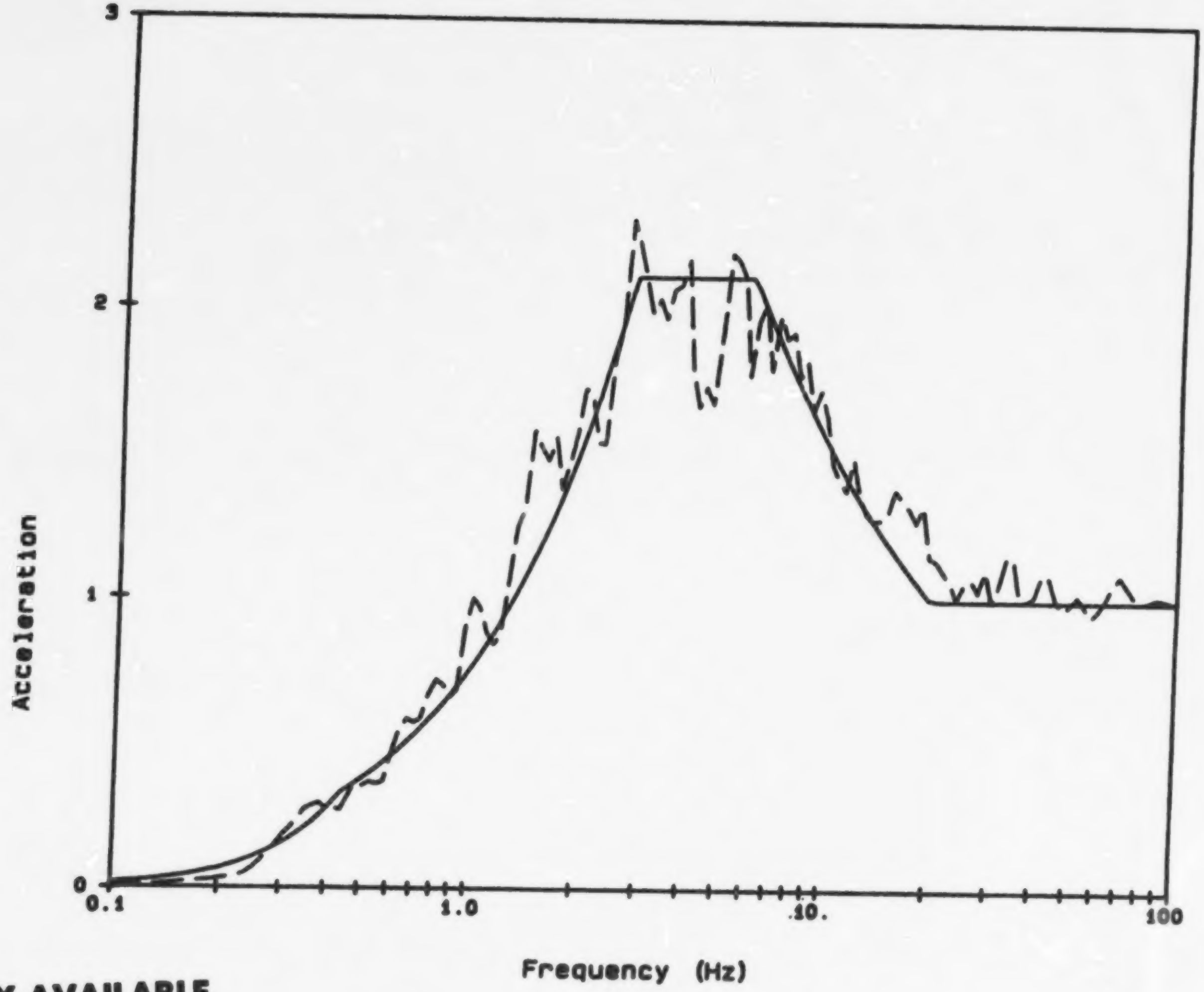

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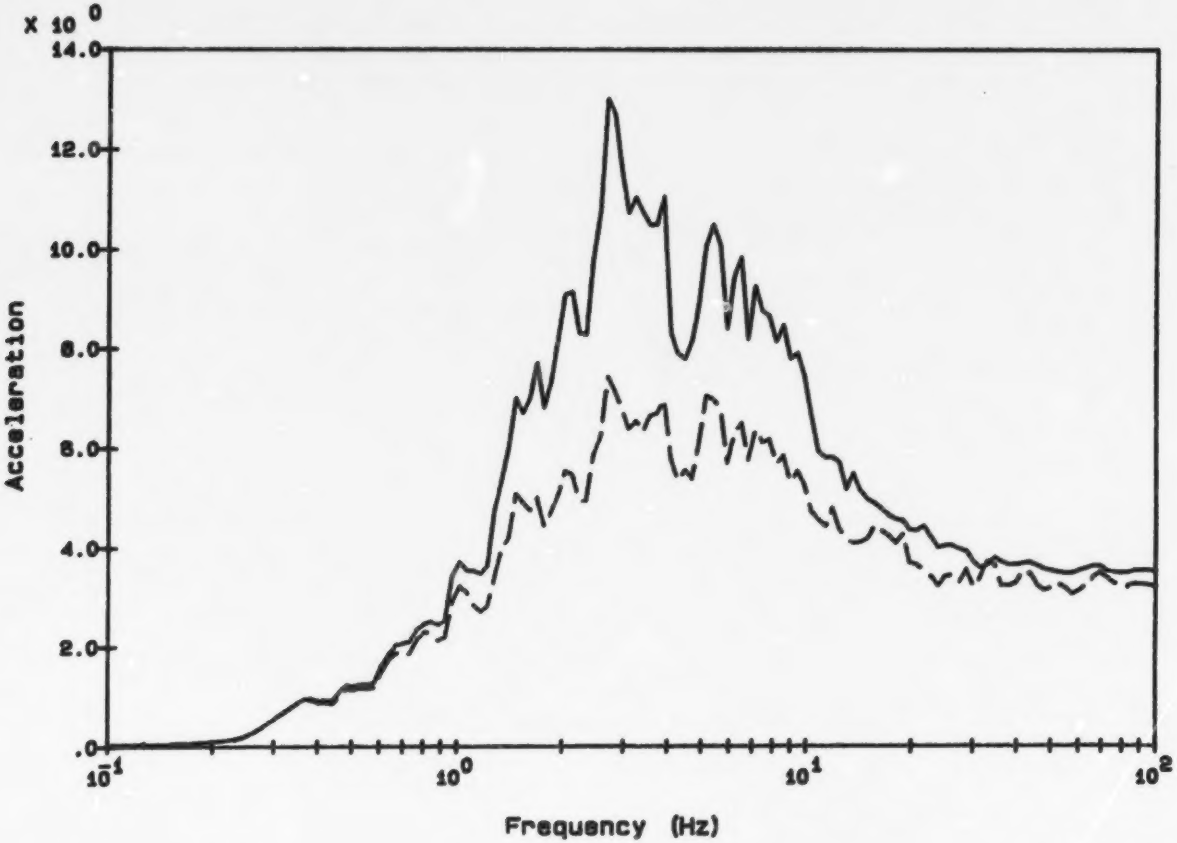
Fig. 5.4 Comparison of Response Spectra on the Outcrop - Median from Ref. 4 vs Artificial Acceleration Time History at 5% Damping



Legend:

Mean of 10 Free field ————

Outcrop Ref. 3 — — — —

Notes:

All accelerations in ft/sec2

All spectra at 5% damping

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Fig 5.5a: Local Site Applification of Outcrop Motion - Acceleration Range 1

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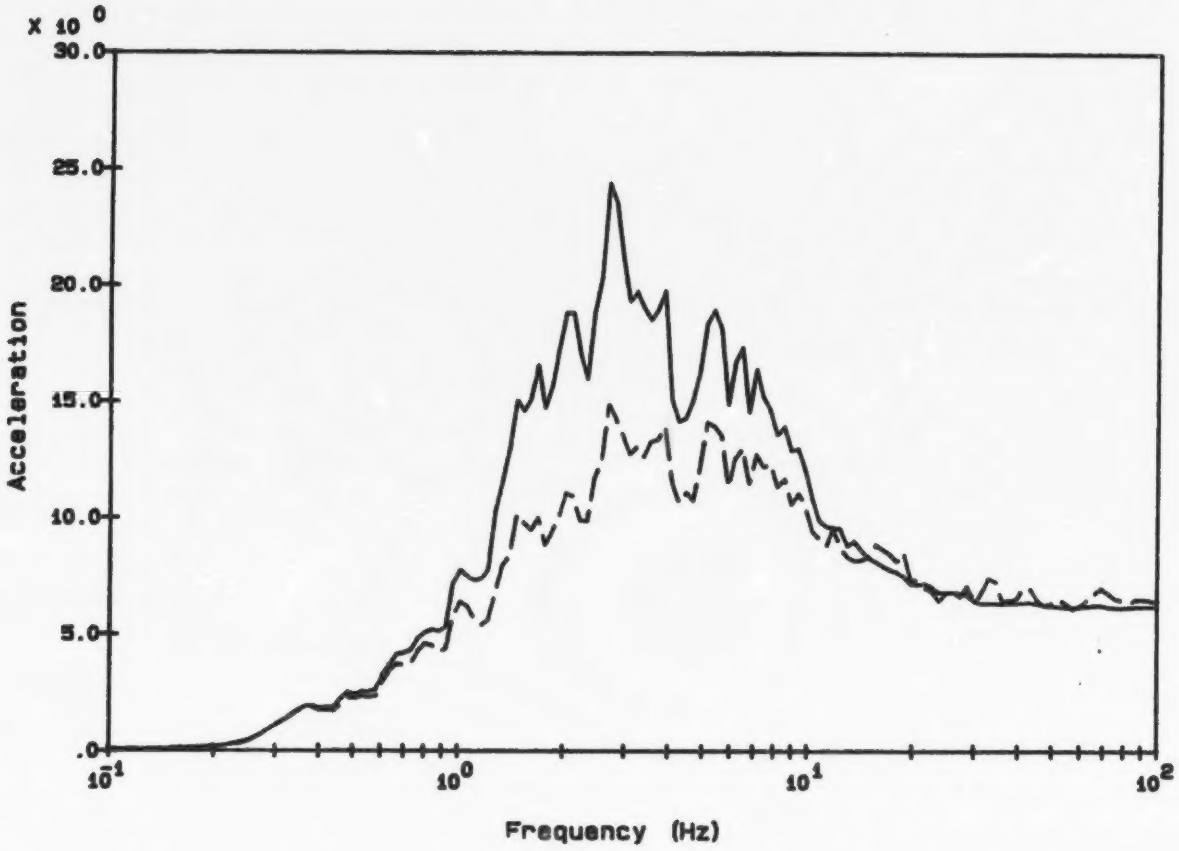

Legend:

Mean of 10 Free field ————————

Outcrop Ref. 3 — — — — .

Notes:

All accelerations in ft/sec2

All spectra at 5% damping

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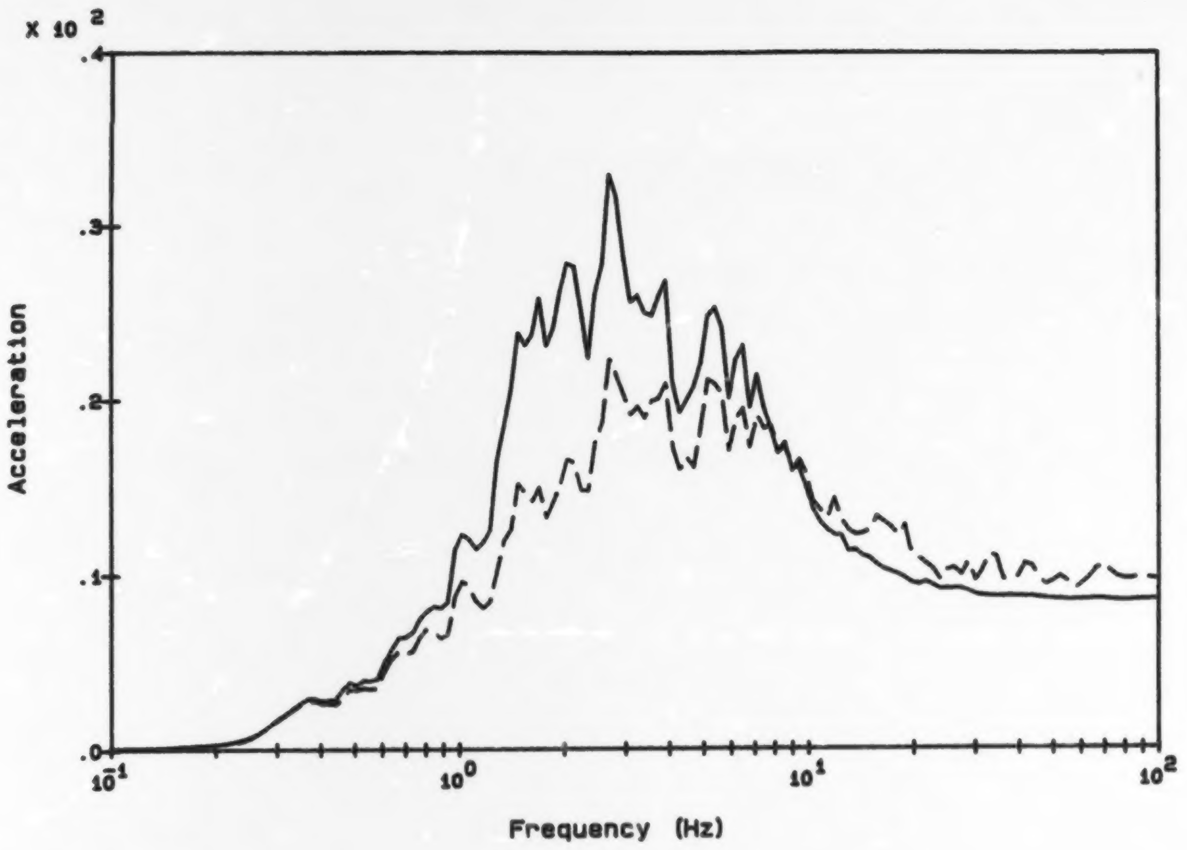
Fig 5.5b: Local Site Applification of Outcrop Motion - Acceleration Range 2



Legend:

Mean of 10 Free field	———
Outcrop Ref. 3	— — — —

Notes:

All accelerations in ft/sec2

All spectra at 5% damping

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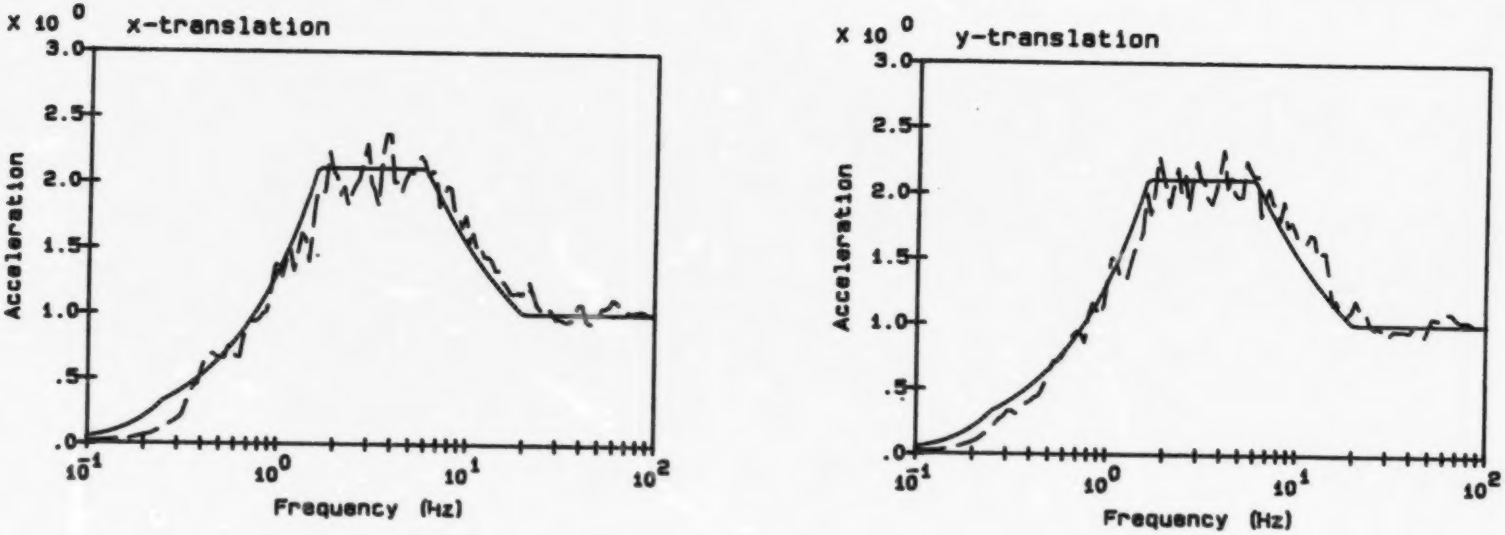
Fig 5.5c: Local Site Applification of Outcrop Motion - Acceleration Range 3



Legend:

Deep alluvium site
as given by ref. 3 ————————

Artificial time
history — — — — — — — -

Notes:

All accelerations in units of g.

All spectra at 5% damping.

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Fig. 5.6: Response Spectra from Artificial Time History and as given by Ref.3

The free-field ground motions for Cases 1 and 2 were used in SSI analyses of the RAB and DGB. For Case 1, probabilistic response analyses were performed. For Case 2, a single analysis at each excitation level was performed to partially quantify the effect of local site amplification.

5.3 Probabilistic Response Analysis of the Reactor Auxiliary Building and Diesel Generator Building

In recognition of the importance of the effects of embedment and soil structure interaction, probabilistic response analyses to generate median response for the seismic PRA were performed on the reactor auxiliary building and the diesel generator building for Case 1 described above. The methodology used is that of SMACS [15] as implemented in the computer program CLASSI [16]utilizing the substructure approach. The substructure approach to SSI is composed of the following elements: Specification of the free-field ground motion; calculation of the dynamic characteristics of the structures; determination of the foundation impedances and analysis of the coupled soil structure system. The procedures is to perform a series of deterministic analyses, each simulating an earthquake occurrence, including variability in seismic input, soil-structure interaction, and in the structure representation. By performing many such analyses and by varying the values of several important parameters, one can account for the uncertainty inherent in any deterministic analysis.

Case 1 and Case 2 Analyses are described:

Case 1: Probabilistic response analyses were performed for Case 1. The ensemble of acceleration time histories generated from the free-field analyses defines the seismic input. Variability in the soil-structure interaction and structure response elements was introduced through a limited number of parameters--soil shear modulus, soil damping, structure frequency and modal damping. Variability in SSI was incorporated through modeling soil shear modulus and material damping as random variables with lognormal distributions -- medians corresponding to the nominal values of Table 5.1 and coefficients of variation of 0.4 and 0.5, respectively. Variability in structure dynamic behavior was also modeled by treating structure frequency and modal damping as random variables. Nominal values of structure frequency were those calculated in the eigenvalue extraction analysis for the structure model. Nominal values of structure damping were taken to be 0.07, 0.085, and 0.10 (fractions of critical damping) for the three seismic acceleration ranges considered here. These were based on published damping values and assumed stress levels achieved.

Parameter variations in each step of the response analysis were selected to represent random variability only. The assumed parameter variability corresponds to that of the SSMRP [3]. The parameter values for each of the ten simulation were selected from the probability distributions by dividing the distributions into equally probable segments, sampling from each segment, and combining the samples by a Latin hypercube experimental design.

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Table 5.1a: Nominal Soil Properties for St. Lucie Analyses-Case 1

	Acceleration Range 1		Acceleration Range 2		Acceleration Range 3	
	RAB	DGB	RAB	DGB	RAB	DGB
Shear Modulus [lb/ft^2]	2.88E6	2.217E6	2.21E6	1.68E6	1.69E6	1.37E6
Damping	.054	.057	.087	.086	.119	.11

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The responses calculated from the simulations are combined to estimate median response conditional on the occurrence of an earthquake described by the particular hazard curve parameter, e.g. peak ground acceleration.

Case 2: A single time history analysis was performed for each acceleration range. Nominal (best-estimate) soil (Table 5.1b) and structure properties were used in the analyses. Cases 1 and 2 were compared to help quantify the effect of local site amplification on structure response.

Reactor Auxiliary Building Response Analyses. The reactor auxiliary building is a reinforced concrete building, rectangular in plan (237 ft. x 113 ft.). The building is supported on a 3 ft. thick reinforced concrete mat at elevation -8.0ft. and extends in height to elevation 82.0 ft. Grade is at 18 ft.

The dynamic model of the reactor auxiliary building used in the response analyses is a 3-D lumped mass model documented in Ref. 1. A fixed-base model excluding the soil springs was reconstructed from the given model data. Figure 5.7 shows the model and correlates node points and floor elevations. Response points associated with DHR components of concern are located within the structure. The structure's fixed-base fundamental modes of vibration in each of the horizontal directions are 9.71 Hz (N-S) and 9.96 Hz (E-W) with 86% and 78.5% of the mass participating in each of the horizontal directions, respectively.

For each of the Case 1 ten earthquake simulations, in-structure horizontal acceleration time histories and corresponding response spectra at 5% damping were calculated for each of the reactor auxiliary building model nodes (1-5). Median response spectra for each node and for each of the horizontal component directions were generated. Figures 5.8 present Case 1 mean horizontal in-structure spectra for the three seismic acceleration levels.

For each of the Case 2 earthquake simulations, in-structure horizontal acceleration time histories and corresponding response spectra at 5% damping were calculated for each of the reactor auxiliary building model nodes (1-5). Figures 5.9 present Case 2 horizontal in-structure spectra for the three seismic acceleration levels.

These spectra formed the basis for determining the response of decay heat removal components housed at various elevation in the reactor auxiliary building.

Diesel Generator Building Response Analyses. The diesel generator building is reinforced concrete building rectangular in plan (80.75 ft. x 60.5 ft). The building is supported on 3 ft. thick mat foundation at elevation 10 ft. extending in height to elevation 49 ft.

The dynamic model of the diesel generator building is a 3-D lumped mass stick model documented in Ref. 8. As for the reactor auxiliary

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Table 5.1b: Nominal Soil Properties for St. Lucie Analyses-Case 2

	Acceleration Range 1		Acceleration Range 2		Acceleration Range 3	
	RAB	DGB	RAB	DGB	RAB	DGB
Shear Modulus [lb/ft^2]	2.75E6	2.151E6	2.08E6	1.65E6	1.66E6	1.32E6
Damping	.061	.056	.093	.088	.12	.114

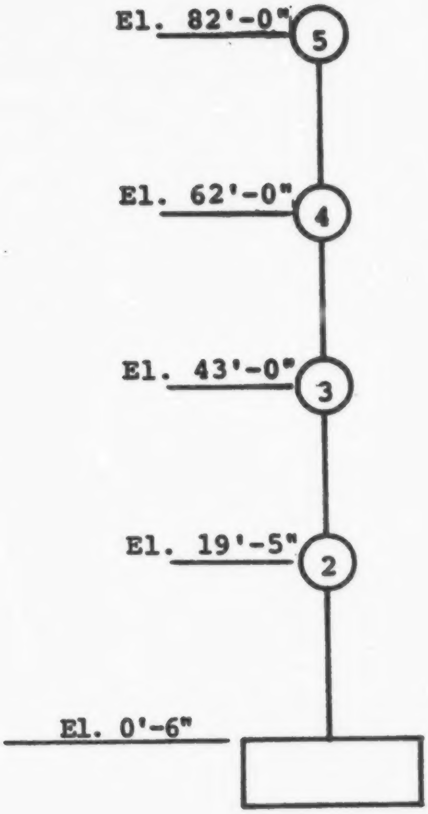
El. 82'-0"
5
El. 62'-0"
4
El. 43'-0"
3
El. 19'-5"
2
El. 0'-6"

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Fig. 5.7 St. Lucie Nuclear Power Plant Reactor Auxiliary Building Structural Model.

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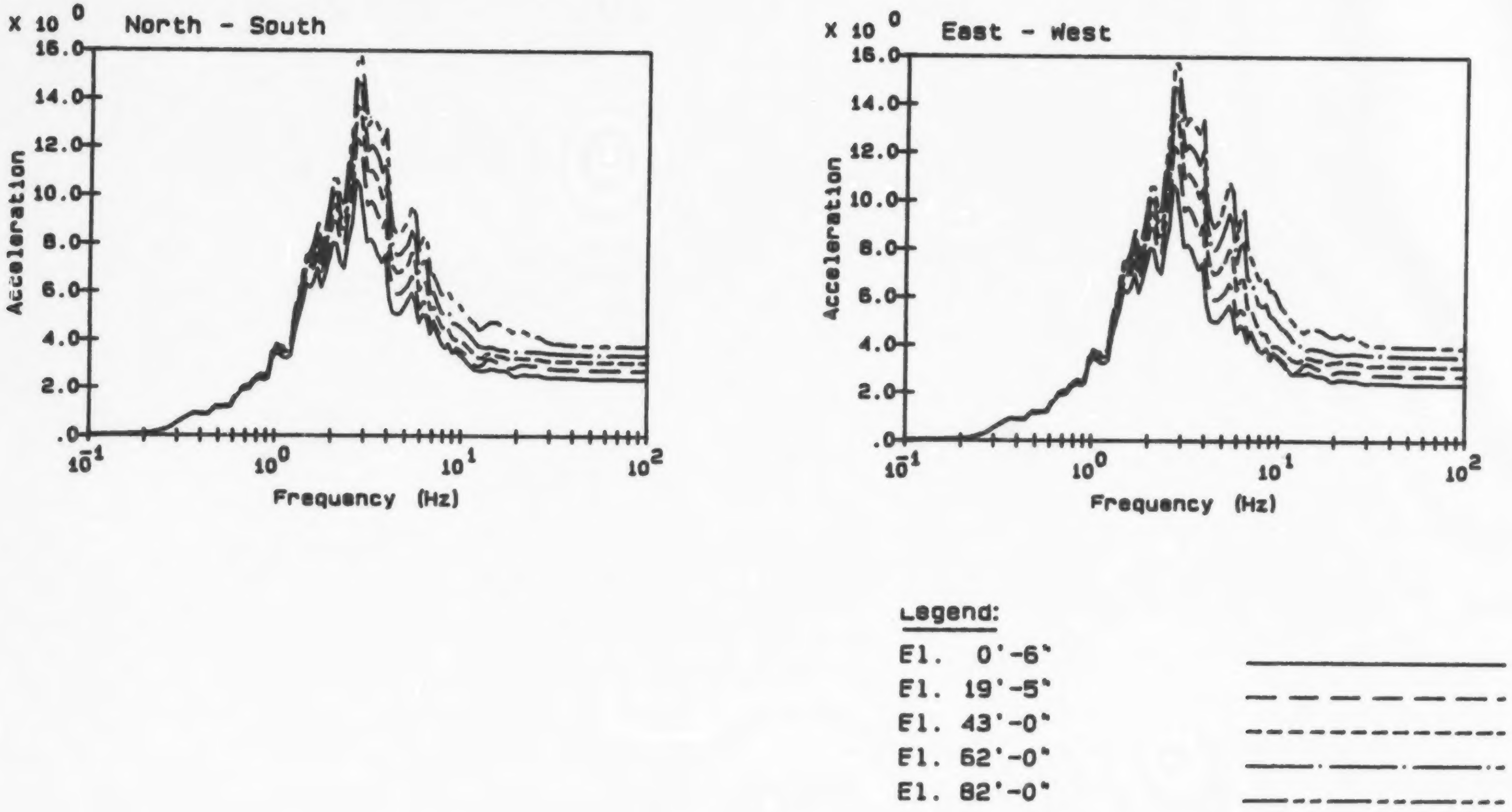

Notes:

All accelerations in units of ft/(s*s).

All spectra calculated at 5% damping.

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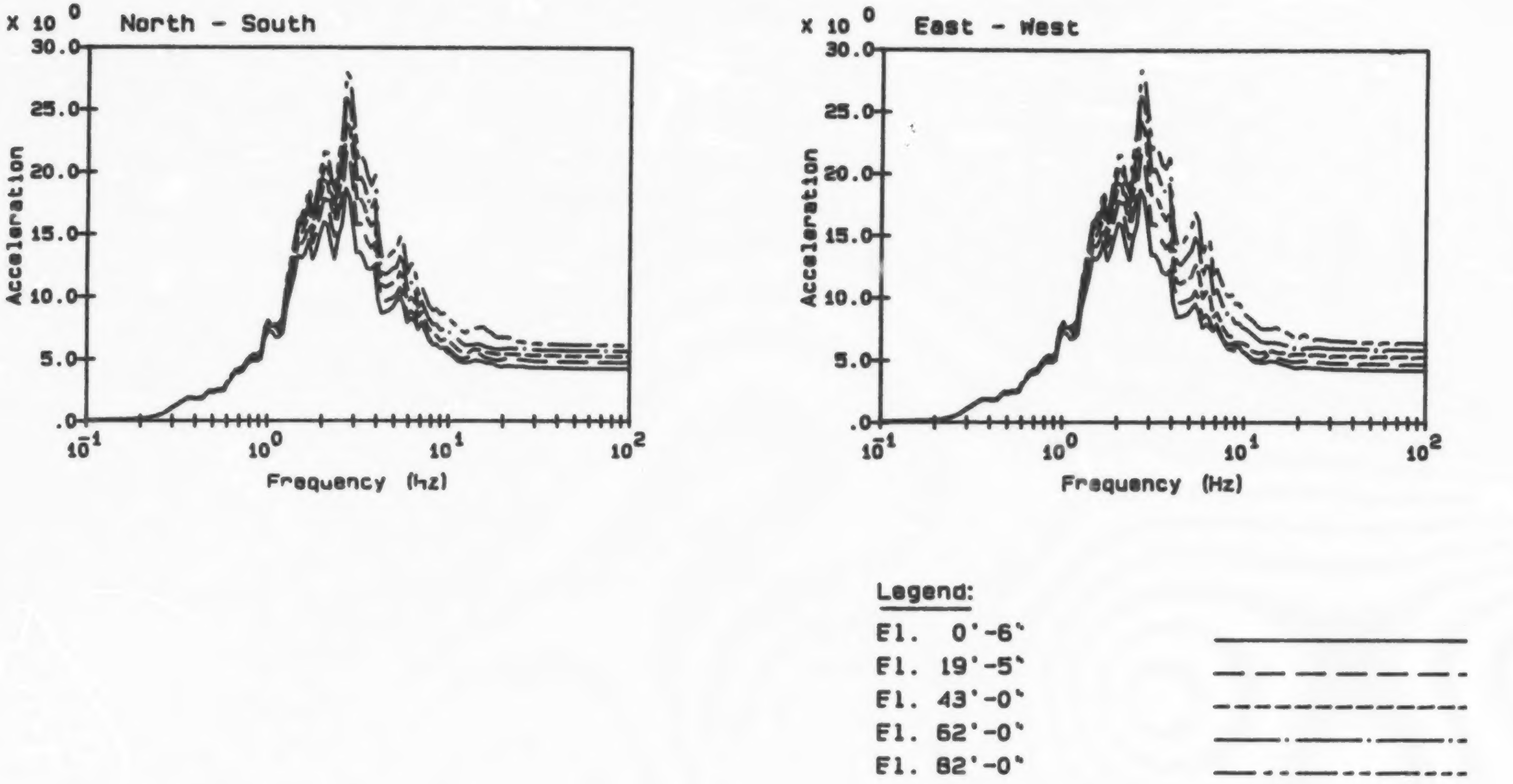
Fig. 5.8a: Reactor Auxiliary Building Mean Instructure Response Acceleration Range 1 - Case 1



Notes:

All accelerations in units of ft/(s*s).
All spectra calculated at 5% damping.

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Fig. 5.8b: Reactor Auxiliary Building Mean Instructure Response Acceleration Range 2 - Case 1

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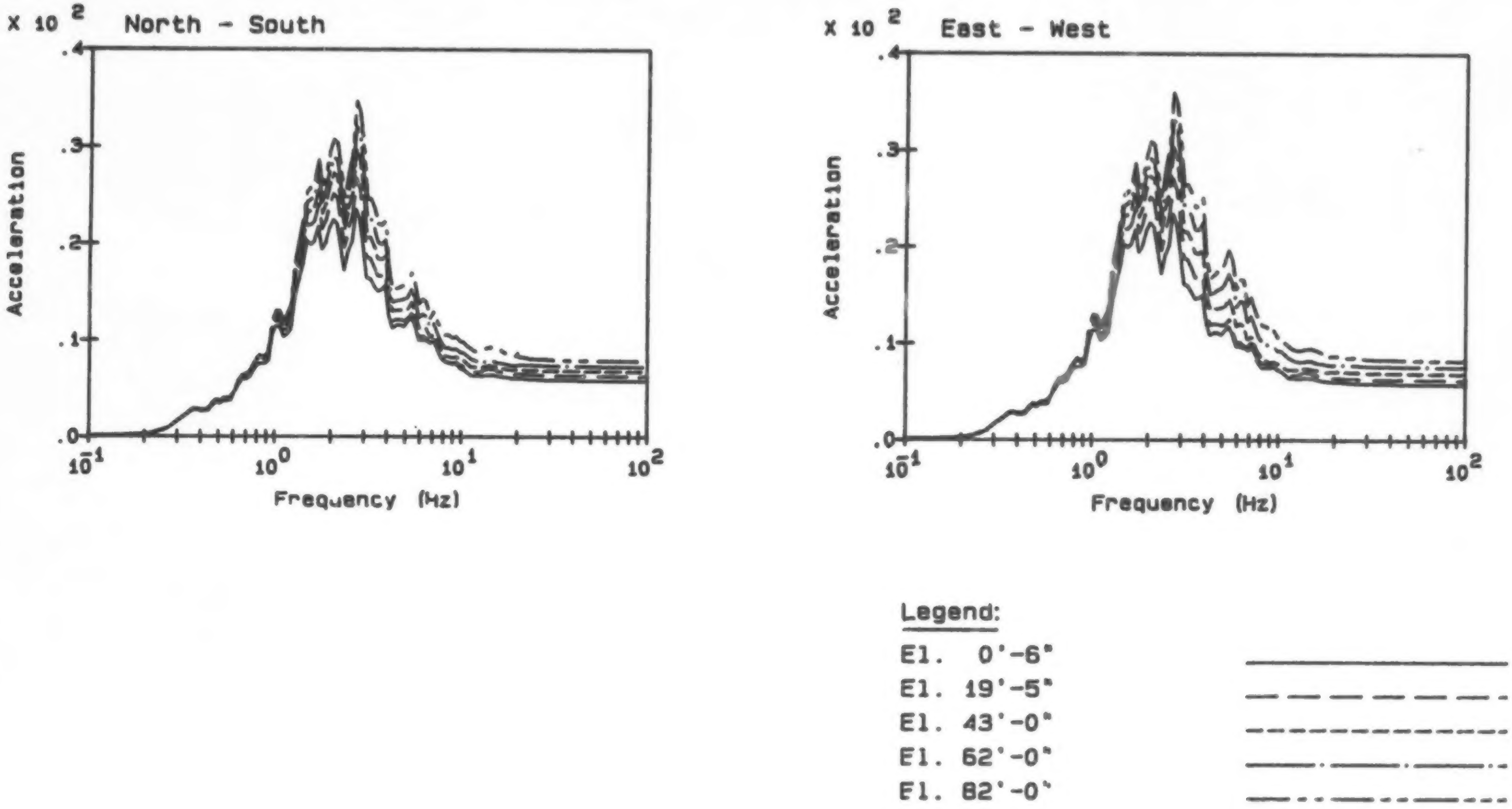

Notes:

All accelerations in units of ft/(s*s).
All spectra calculated at 5% damping.

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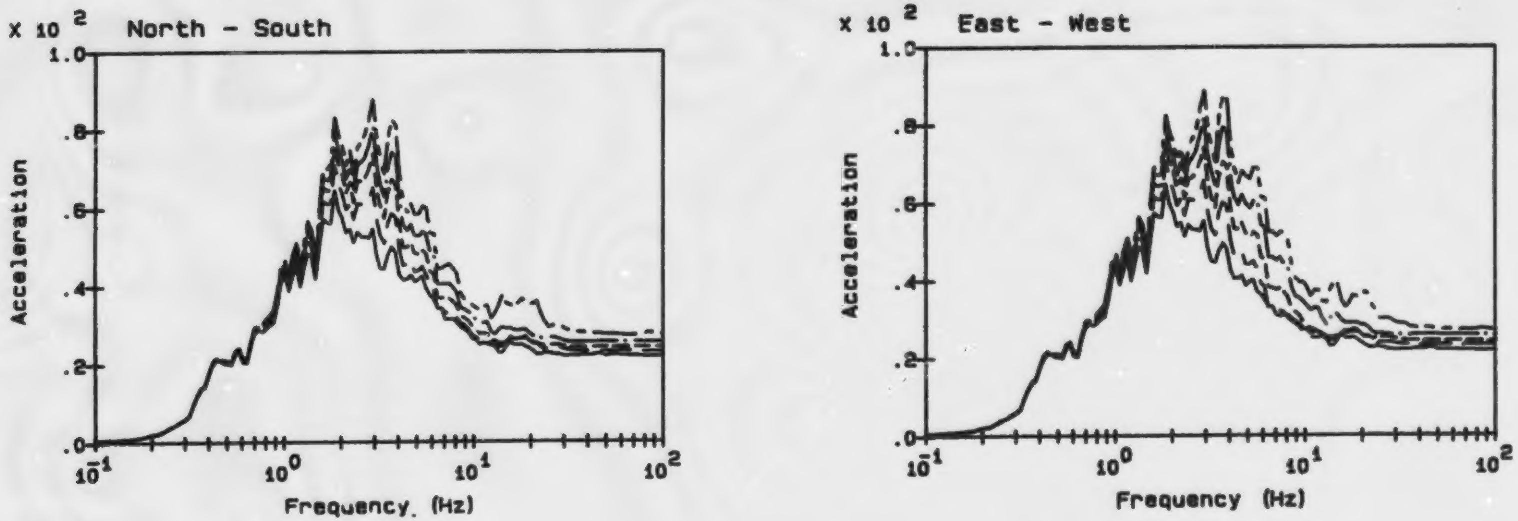
Fig. 5.8c: Reactor Auxiliary Building Mean Instructure Response
Acceleration Range 3 - Case 1



Legend:

El. 0'-6"	———————
El. 19'-5"	— — — — — — —
El. 43'-0"	- - - - - - - - - - - -
El. 62'-0"	——·——·——·
El. 82'-0"	——- - -——- - -

Notes:

All accelerations in units of ft/(s*s).
All spectra calculated at 5% damping.

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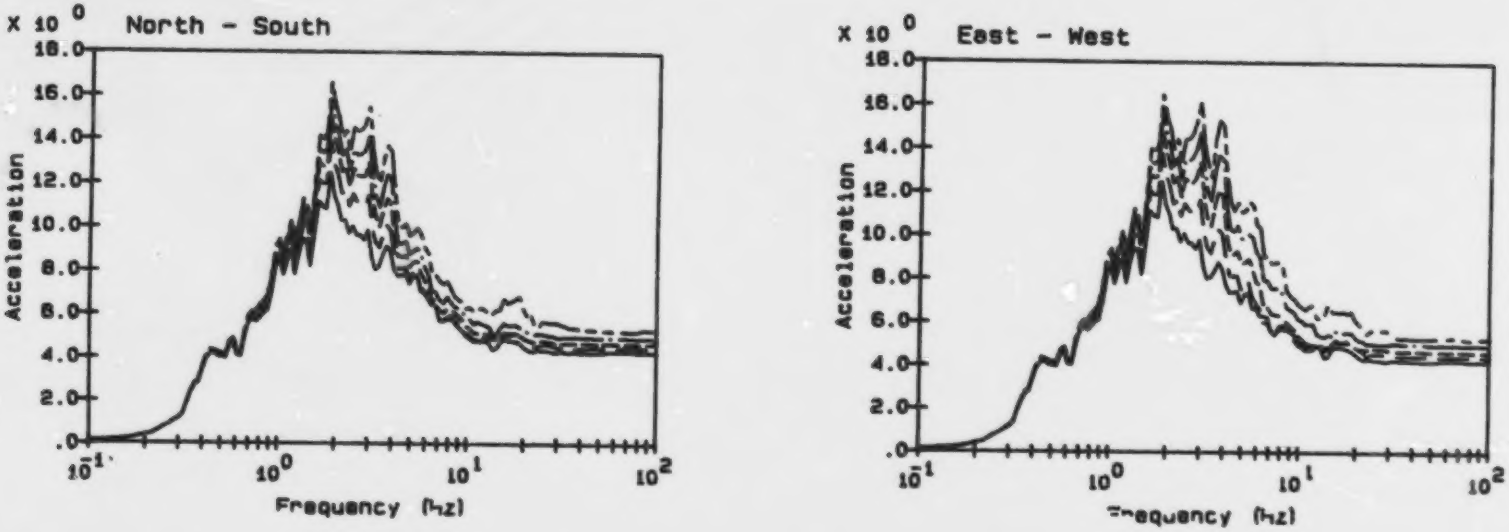
Fig. 5.9a: Reactor Auxiliary Building Instructure Response, Acceleration Range 1 - Case 2



Legend:

El. 0'-6"	————————
El. 19'-5"	— — — — — — -
El. 43'-0"	------------
El. 62'-0"	——·——·——·-
El. 82'-0"	——-——-——---

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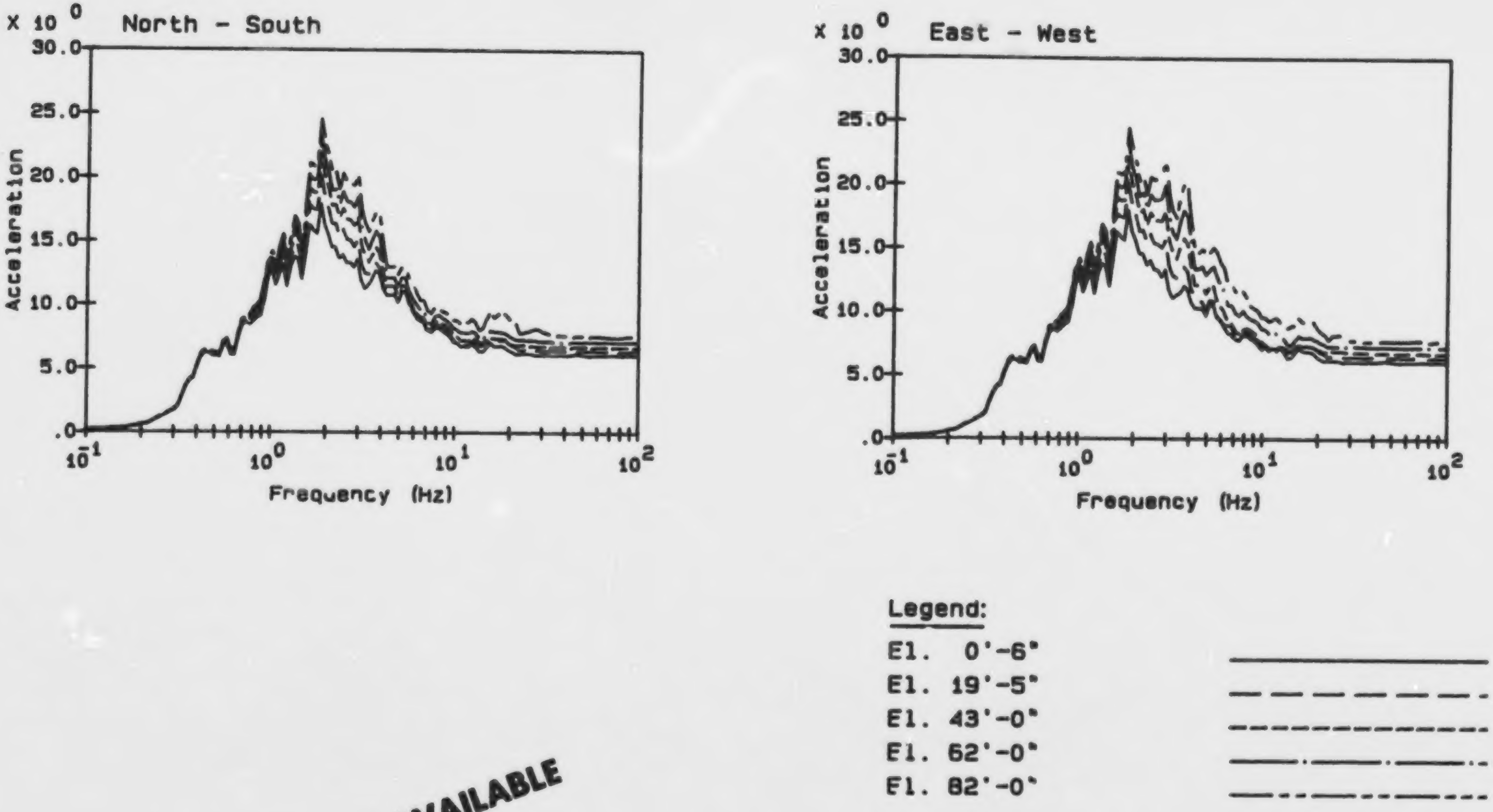
Notes:

All accelerations in units of ft/(s*s).
All spectra calculated at 5% damping.

Fig. 5.9b: Reactor Auxiliary Building Instructure Response,
Acceleration Range 2 - Case 2



Notes:

All accelerations in units of ft/(s*s).

All spectra calculated at 5% damping.

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Fig. 5.9c: Reactor Auxiliary Building Instructure Response, Acceleration Range 3 - Case 2

building, a fixed-base model of the diesel generator building excluding the soil springs was reconstructed from the given data and used for the probabilistic response analyses. Figure 5.10 shows the model and correlates node points and floor elevations. The model as shown in Figure 5.10 consists of three cantilever beams; one representing the structure (Nodes 1, 4-6) and two representing the diesel generators (Nodes 2,3) and their foundations. All three cantilever beams are supported on the foundation mat. The diesel generator building fixed-base fundamental modes of vibration have a frequency of 18.7 Hz (N-S) and 25.2 Hz (E-W) with 52.8% and 52.4% mass participation, respectively.

For each of the Case 1 ten earthquake simulations in-structure horizontal acceleration time histories and corresponding response spectra at 5% damping were calculated for each of the diesel generator building model nodes (1-6). These spectra were then combined into median responses for each node in diesel generator building model and for each of the horizontal component directions. Figures 5.11 present Case 1 mean horizontal in-structure spectra for three seismic acceleration levels.

For each of the Case 2 earthquake simulations, in-structure horizontal acceleration time histories and corresponding response spectra at 5% damping were calculated at each of the diesel generator auxiliary building model nodes (1-6). Figure 5.12 presents Case 2 horizontal in-structure spectra for the three seismic acceleration levels.

These spectra formed the basis for determining the response of decay heat removal components housed at various elevations in the diesel generator building.

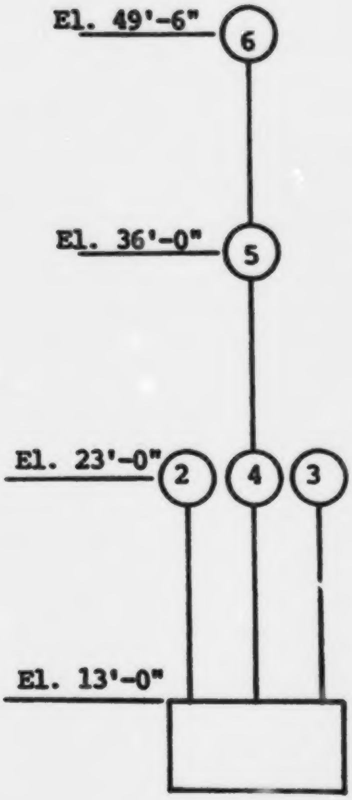
El. 49'-6"
6
El. 36'-0"
5
El. 23'-0"
2
4
3
El. 13'-0"

Fig. 5.10 St. Lucie Nuclear Power Plant Diesel Generator Building Structural Model

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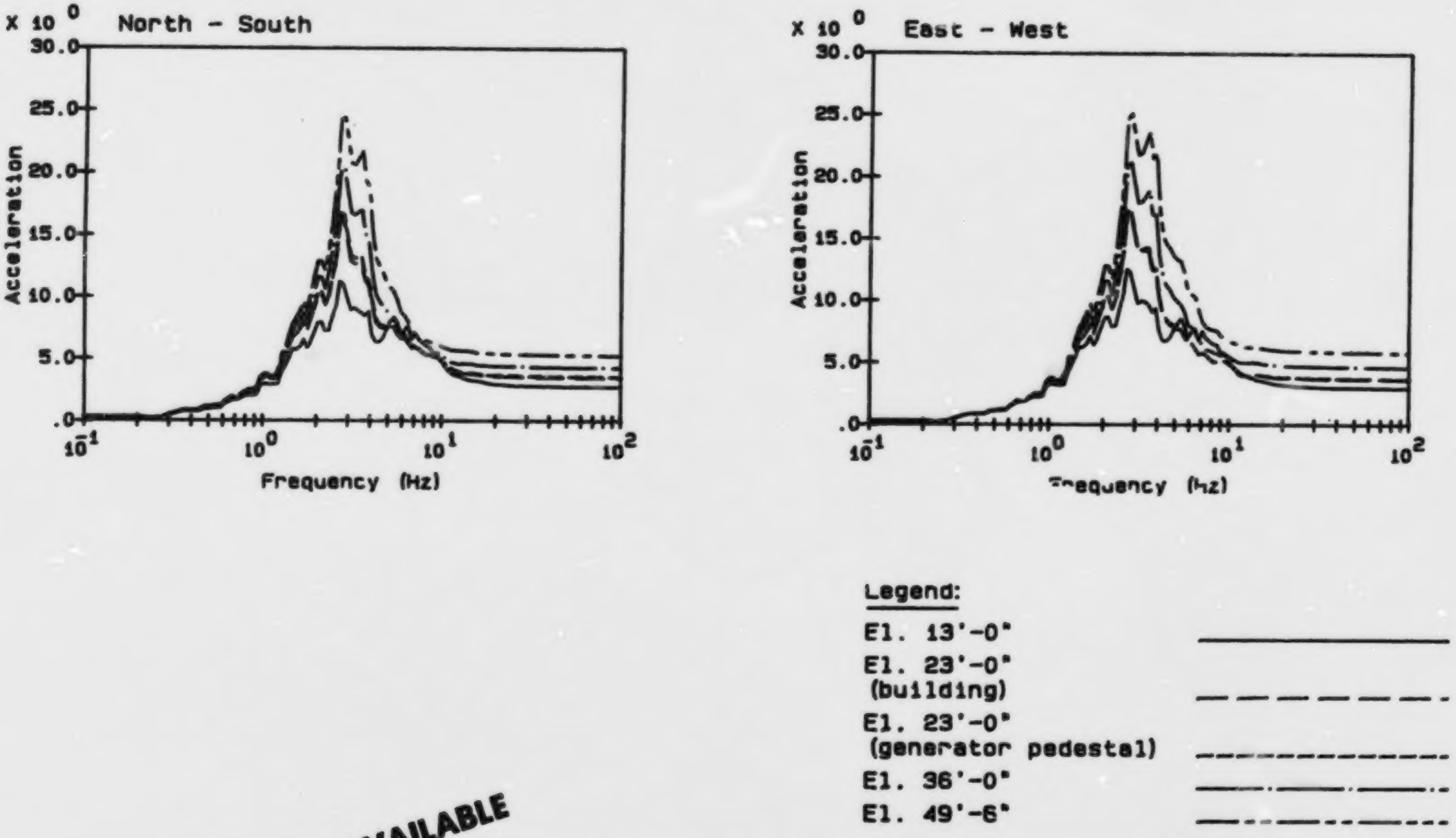

Notes:

All accelerations in units of ft/(s*s).

All spectra calculated at 5% damping.

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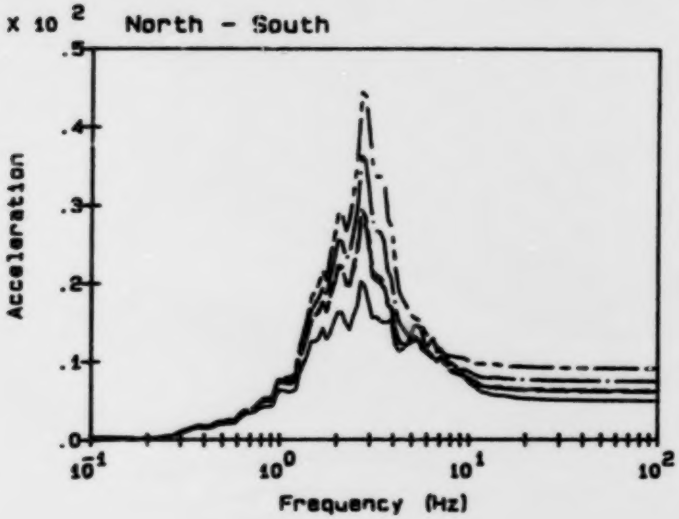
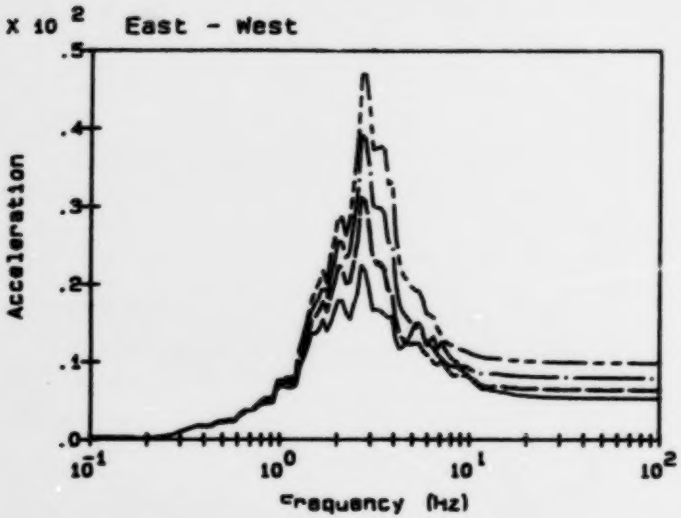
Fig. 5.11a: Diesel Generatory Building Mean Instructure Response
Acceleration Range 1 - Case 1





Legend:

El. 13'-0" ————————

El. 23'-0" (building) — — — — — — —

El. 23'-0" (generator pedestal) - - - - - - - - - - - -

El. 36'-0" ——·——·——·—

El. 49'-6" ——-·-·——-·-·—

Notes:

All accelerations in units of ft/(s*s).

All spectra calculated at 5% damping.

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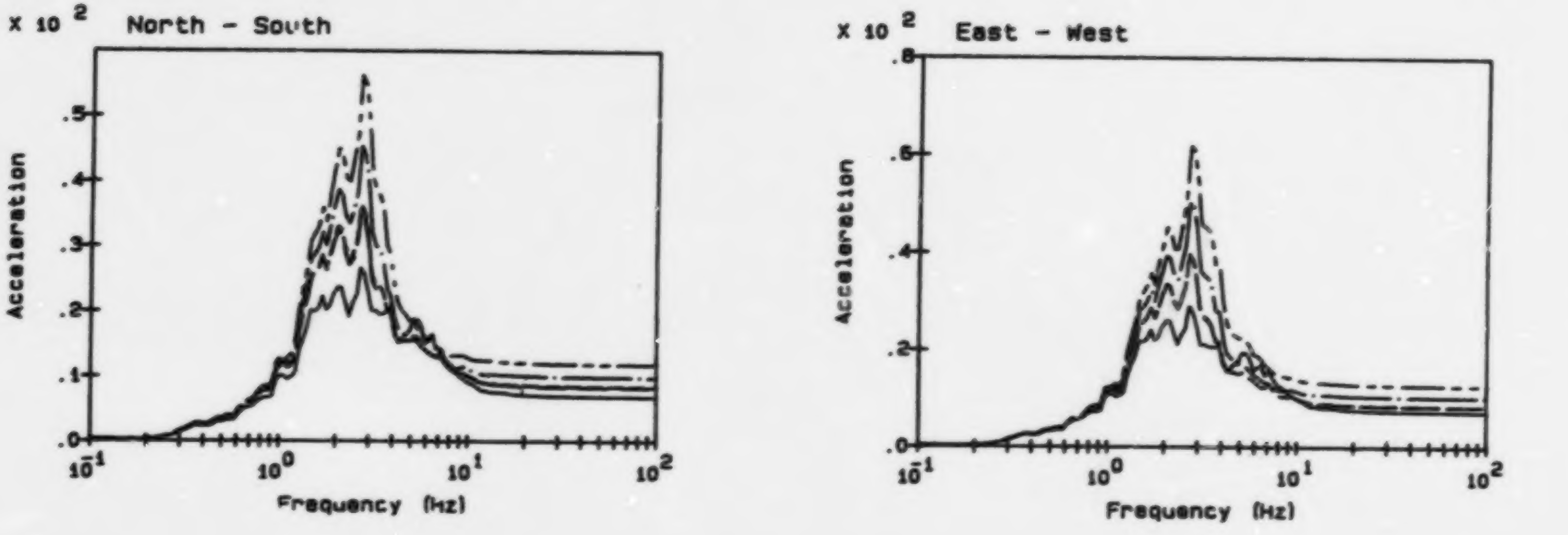
Fig. 5.11b: Diesel Generatory Building Mean Instructure Response Acceleration Range 2 - Case 1



Legend:

El. 13'-0" ————————
El. 23'-0" (building) — — — — — — .
El. 23'-0" (generator pedestal) ------------.
El. 36'-0" ——·——·——·-
El. 49'-6" ——--——--——---

Notes:

All accelerations in units of ft/(s*s).

All spectra calculated at 5% damping.

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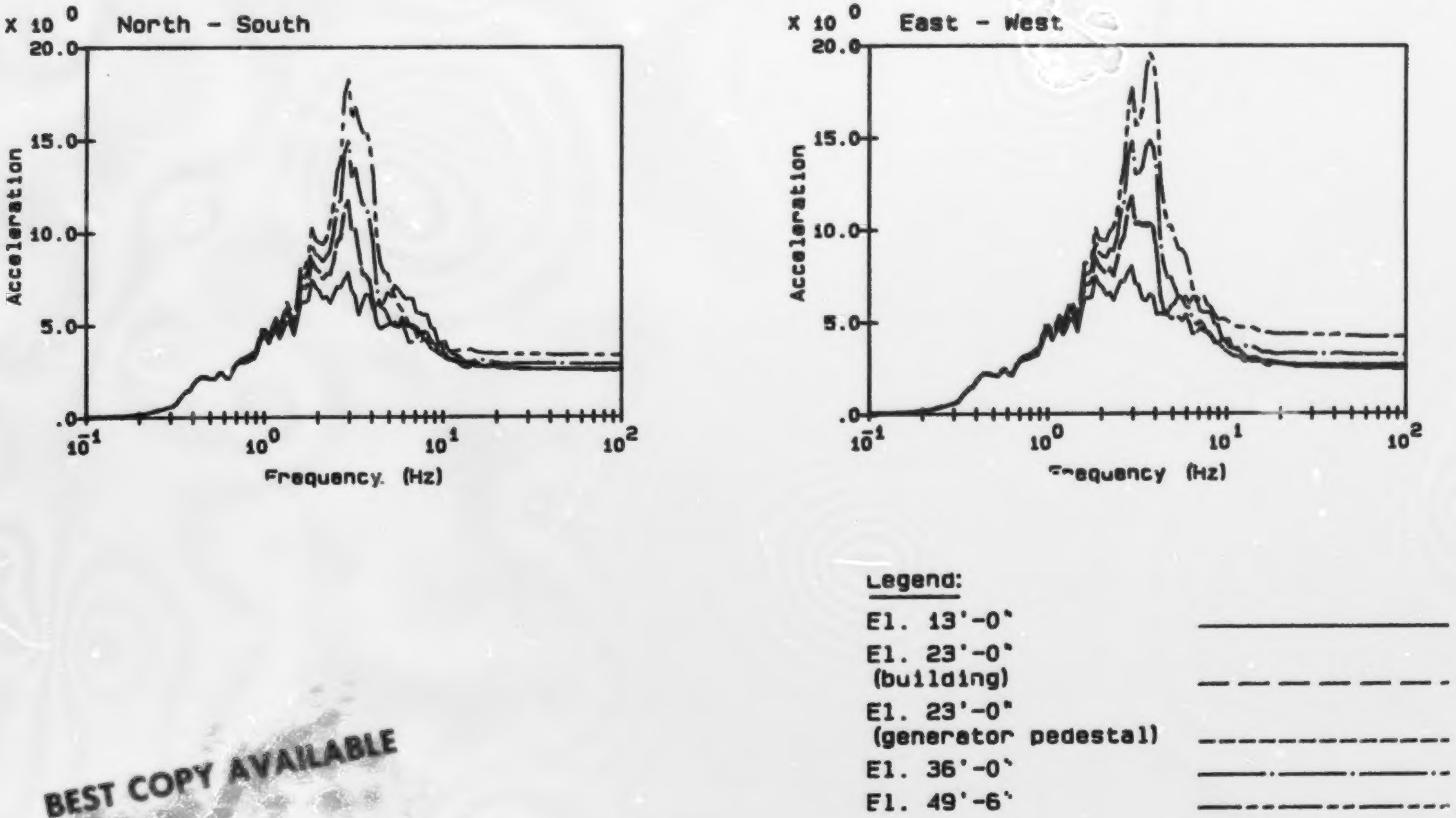
Fig. 5.11c: Diesel Generatory Building Mean Instructure Response
Acceleration Range 3 - Case 1



Notes:

All accelerations in units of ft/(s*s).

All spectra calculated at 5% damping.

Fig. 5.12a: Diesel Generator Building Instructure Response,
Acceleration Range 1 - Case 2

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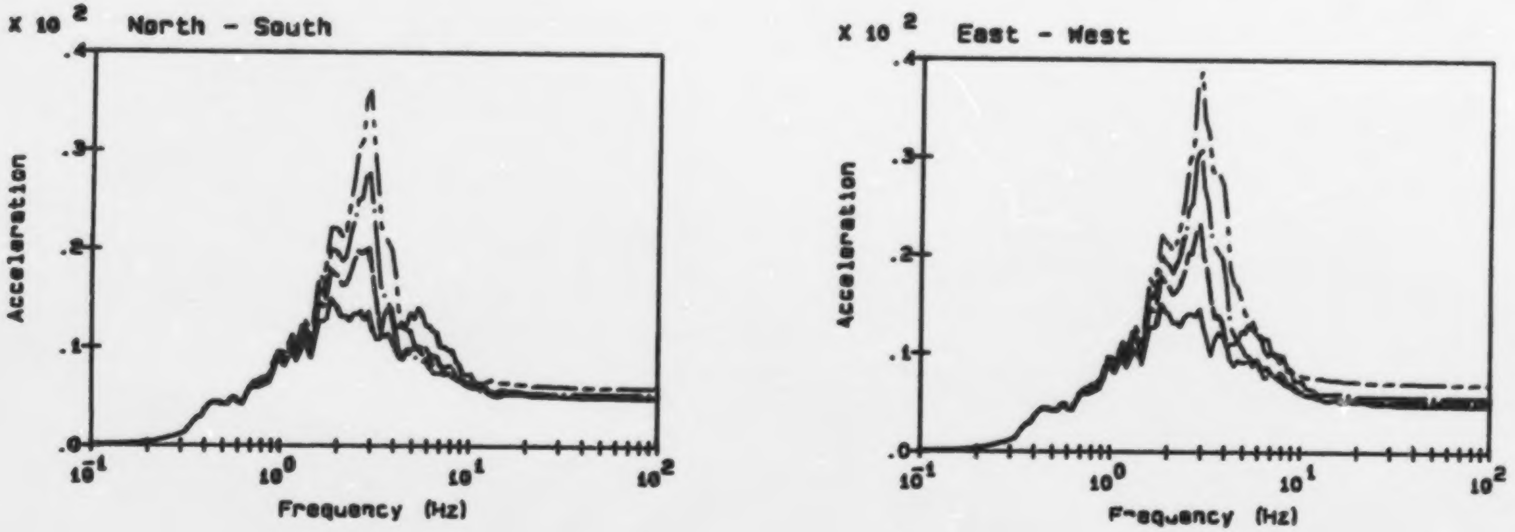

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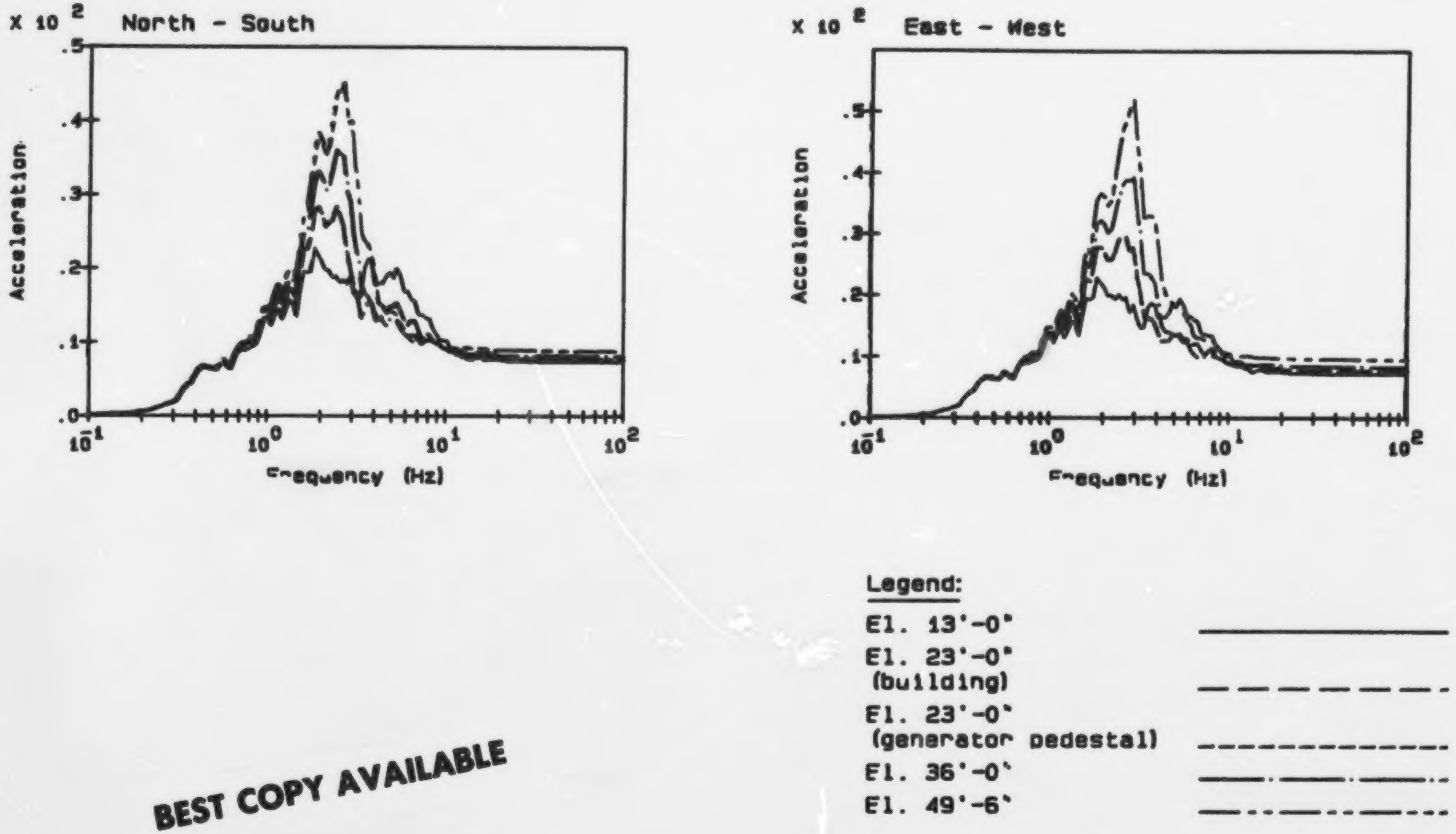
Legend:

El. 13'-0" ————————
El. 23'-0" (building) — — — — — —
El. 23'-0" (generator pedestal) - - - - - - - - - -
El. 36'-0" ——·——·——·
El. 49'-6" ——··——··——··

Notes:

All accelerations in units of ft/(s*s).
All spectra calculated at 5% damping.

Fig. 5.12b: Diesel Generator Building Instructure Response, Acceleration Range 2 - Case 2



Notes:

All accelerations in units of ft/(s*s).

All spectra calculated at 5% damping.

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Fig. 5.12c: Diesel Generator Building Instructure Response,
Acceleration Range 3 - Case 2

5.4 Decay Heat Removal (DHR) System Component Responses

The median in-structure spectra presented and discussed in the previous section are used to determine DHR component response. Assuming that the dynamic characteristics of a given component can be represented by a single dominant mode of vibration, the component response can be approximated by the spectral acceleration of the appropriate in-structure spectra at the frequency of the dominant mode.

For the three levels of ground motion considered in this study, best estimate DHR systems component responses are presented in Table 5.2 for Case 1 and Table 5.3 for Case 2. At each structural location, response is calculated over several frequency ranges. These ranges span the probable natural frequencies of the components housed at that location. The median zero period acceleration response is calculated from the ten values given by the probabilistic response anlaysis. The median response over a frequency range is calculated by determining the spectral accelerations at 25 points over the range from the spectra given by the ten earthquake simulations. These spectral accelerations in turn are used to determine the median response. Given the natural frequency of the component of interest, the appropriate frequency interval and component response is then defined.

Neither Case 1 nor Case 2 totally represent the St., Lucie scenario; hence some combination of responses should be used. Enveloping the two, and acknowledging this may be conservative may be appropriate.

Table 5.2 St. Lucie Seismic Responses - Case 1

Response for Given Peak Ground Acceleration (G's).

			0.10		0.20		0.30	
Location	**Elevation**	**Frequency**	**X-Trans**	**Y-Trans**	**X-Trans**	**Y-Trans**	**X-Trans**	**Y-Trans**
RAB	-.5	2-5	.22	.22	.38	.37	.48	.47
RAB	-.5	5-10	.13	.13	.22	.22	.27	.27
RAB	-.5	10-20	.08	.09	.14	.15	.19	.19
RAB	-.5	ZPA	.07	.07	.13	.13	.18	.18
RAB	19.5	2-5	.25	.25	.43	.43	.52	.53
RAB	19.5	5-10	.14	.14	.22	.22	.27	.27
RAB	19.5	10-20	.09	.09	.15	.15	.20	.20
RAB	19.5	ZPA	.08	.08	.14	.14	.19	.19
RAB	43.0	2-5	.28	.29	.47	.48	.57	.59
RAB	43.0	5-10	.16	.16	.24	.25	.29	.30
RAB	43.0	10-20	.10	.10	.16	.17	.21	.22
RAB	43.0	ZPA	.09	.09	.15	.16	.20	.20
RAB	62.0	2-5	.31	.32	.51	.53	.61	.65
RAB	62.0	5-10	.17	.20	.27	.30	.31	.34
RAB	62.0	10-20	.11	.12	.18	.19	.23	.24
RAB	62.0	ZPA	.10	.11	.17	.17	.21	.22
RAB	82.0	2-5	.33	.35	.55	.58	.66	.71
RAB	82.0	5-10	.20	.23	.30	.34	.34	.39
RAB	82.0	10-20	.13	.14	.21	.22	.25	.27
RAB	82.0	ZPA	.11	.12	.18	.19	.23	.24

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Table 5.2 St. Lucie Seismic Responses - Case 1 (Continued)

Response for Given Peak Ground Acceleration (G's).

			0.10		0.20		0.30	
Location	**Elevation**	**Frequency**	**X-Trans**	**Y-Trans**	**X-Trans**	**Y-Trans**	**X-Trans**	**Y-Trans**
DGB	13.0	2-5	.24	.27	.42	.49	.56	.64
DGB	13.0	5-10	.19	.21	.30	.35	.37	.43
DGB	13.0	10-20	.10	.11	.16	.19	.21	.25
DGB	13.0	ZPA	.08	.09	.14	.16	.20	.22
DGB	23.0*	2-5	.32	.35	.53	.59	.67	.72
DGB	23.0*	5-10	.18	.18	.30	.29	.38	.36
DGB	23.0*	10-20	.12	.12	.20	.21	.27	.27
DGB	23.0*	ZPA	.10	.11	.18	.19	.25	.25
DGB	23.0**	2-5	.33	.35	.56	.59	.69	.73
DGB	23.0**	5-10	.18	.18	.30	.29	.37	.36
DGB	23.0**	10-20	.12	.12	.21	.21	.27	.27
DGB	23.0**	ZPA	.11	.11	.19	.19	.25	.26
DGB	36.0	2-5	.40	.45	.66	.74	.79	.90
DGB	36.0	5-10	.19	.20	.30	.32	.38	.39
DGB	36.0	10-20	.14	.15	.24	.25	.31	.32
DGB	36.0	ZPA	.13	.14	.23	.24	.30	.31
DGB	49.5	2-5	.50	.55	.82	.93	.98	1.13
DGB	49.5	5-10	.22	.27	.35	.47	.43	.48
DGB	49.5	10-20	.17	.19	.29	.31	.38	.39
DGB	49.5	ZPA	.16	.18	.28	.30	.36	.38

*Node 2 **Node 4

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Table 5.2 St. Lucie Seismic Responses - Case 1 (Continued)

Response for Given Peak Ground Acceleration (G's).

			0.10		0.20		0.30	
Location	Elevation	Frequency	X-Trans	Y-Trans	X-Trans	Y-Trans	X-Trans	Y-Trans
YARD	18.	2-5	.30	.30	.55	.55	.75	.75
YARD	18.	5-10	.27	.27	.46	.46	.58	.58
YARD	18.	10-20	.16	.16	.26	.26	.33	.33
YARD	18.	ZPA	.11	.11	.19	.19	.26	.26

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Table 5.3: St. Lucie Seismic Responses - Case 2

			Response for Given Peak Ground Acceleration (G's)					
			0.10		0.20		0.30	
Location	Elevation	Frequency	X-Trans.	Y-Trans.	X-Trans.	Y-Trans.	X-Trans.	Y-Trans.
RAB	-.5	2-5	.15	.15	.27	.27	.38	.37
RAB	-.5	5-10	.10	.10	.19	.19	.27	.26
RAB	-.5	10-20	.08	.08	.15	.15	.21	.21
RAB	-.5	ZPA	.07	.07	.13	.13	.19	.19
RAB	19.5	2-5	.17	.17	.30	.31	.41	.41
RAB	19.5	5-10	.11	.10	.19	.19	.26	.26
RAB	19.5	10-20	.08	.08	.15	.15	.22	.22
RAB	19.5	ZPA	.07	.07	.14	.14	.20	.20
RAB	43.0	2-5	.19	.19	.33	.35	.44	.47
RAB	43.0	5-10	.12	.12	.20	.21	.28	.28
RAB	43.0	10-20	.08	.08	.16	.16	.23	.22
RAB	43.0	ZPA	.08	.08	.14	.15	.21	.21
RAB	62.0	2-5	.20	.21	.36	.38	.48	.52
RAB	62.0	5-10	.13	.14	.22	.24	.29	.32
RAB	62.0	10-20	.09	.09	.17	.17	.24	.25
RAB	62.0	ZPA	.08	.08	.15	.15	.22	.22
RAB	82.0	2-5	.22	.23	.39	.42	.52	.57
RAB	82.0	5-10	.14	.16	.24	.28	.32	.38
RAB	82.0	10-20	.11	.11	.20	.20	.28	.28
RAB	82.0	ZPA	.09	.08	.16	.16	.23	.24

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Table 5.3, St Lucie Seismic Responses - Case 2, (Continued)

Location	Elevation	Frequency	Response for Given Peak Ground Acceleration (G's) 0.10 X-Trans.	0.10 Y-Trans.	0.20 X-Trans.	0.20 Y-Trans.	0.30 X-Trans.	0.30 Y-Trans.
DGB	13.0	2-5	.20	.20	.40	.40	.58	.55
DGB	13.0	5-10	.17	.17	.32	.31	.44	.44
DGB	13.0	10-20	.09	.09	.18	.17	.25	.25
DGB	13.0	ZPA	.08	.08	.15	.15	.23	.22
DGB	23.0 *	2-5	.24	.26	.42	.46	.56	.60
DGB	23.0 *	5-10	.14	.14	.25	.25	.36	.35
DGB	23.0 *	10-20	.09	.09	.17	.17	.25	.25
DGB	23.0 *	ZPA	.08	.08	.16	.16	.24	.24
DGB	23.0**	2-5	.24	.26	.42	.46	.57	.61
DGB	23.0**	5-10	.14	.14	.26	.25	.36	.35
DGB	23.0**	10-20	.09	.09	.17	.17	.25	.25
DGB	23.0**	ZPA	.08	.08	.16	.16	.24	.24
DGB	36.0	2-5	.30	.34	.52	.60	.66	.78
DGB	36.0	5-10	.13	.15	.23	.24	.33	.33
DGB	36.0	10-20	.09	.11	.17	.18	.26	.27
DGB	36.0	ZPA	.09	.10	.17	.17	.25	.26
DGB	49.5	2-5	.39	.43	.67	.78	.84	1.03
DGB	49.5	5-10	.15	.20	.24	.32	.33	.40
DGB	49.5	10-20	.11	.14	.20	.23	.28	.31
DGB	49.5	ZPA	.10	.13	.18	.22	.27	.29

* Node 2 ** Node 4

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Table 5.3: **St Lucie Seismic Responses - Case 2,** (Continued)

			Response for Given Peak Ground Acceleration (G's)					
			0.10		0.20		0.30	
Location	Elevation	Frequency	X-Trans.	Y-Trans.	X-Trans.	Y-Trans.	X-Trans.	Y-Trans.
YARD	18.	2-5	.21	.21	.41	.41	.62	.62
YARD	18.	5-10	.19	.19	.39	.39	.58	.58
YARD	18.	10-20	.13	.14	.27	.27	.40	.41
YARD	18.	ZPA	.10	.10	.20	.20	.30	.30

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5.5 Responses in Terms of Peak Ground Acceleration

The responses in Table 5.2 are given at three rock outcrop accelerations (0.10g, 0.20g and 0.30g). Figures 5.12 and 5.13 are plots of the response point accelerations in each building versus the peak ground acceleration level (Response 1). It can be seen that a linear relation exists up to free-field accelerations of 0.30g or greater. Furthermore, for those curves which show significant non-linearity at higher acceleration levels, the linear relation provides a conservative estimate of the local response. Such behavior was first recognized in the SSMRP studies (Reference 3). Inasmuch as the bulk of the risk is generally due to earthquake levels only up to the 4 SSE level, it can be seen that relating local response to peak ground acceleration in a linear fashion is appropriate for the St. Lucie analysis. From these figures, ratios between the various responses and the surface peak ground acceleration were determined, as listed on Table 5.3. Table 5.4 lists all the basic event component failures, their locations, response numbers and fragility categories. These two tables can be used in conjunction with tables of the generic component fragility values (Appendix A) to obtain the component failure probabilities directly.

5.6 Variability in Response

Variability in responses (floor slab and spectral accelerations) is assigned based on the detailed SSMRP response calculations made for the Zion Nuclear Power Plant (Reference 3). In this evaluation, an attempt was made to accurately compute the responses of walls and floor slabs in the Zion structures, moments in the important piping systems, accelerations of all important valves, and the spectral accelerations at each safety system component (pump, electrical buss, motor control center, etc.). Correlation between the responses of all components was computed directly from the dynamic response calculations. Detailed summary plots showed that the uncertainty in the various responses could be reasonably approximated as follows:

Response Type	β random
Peak Ground Acceleration	0.25
Floor Zero Period Acceleration	0.35
Floor Spectral Acceleration	0.45

where β is the standard deviation of the logarithm of the response. These values were used in computation all the generic component failure probabilities in Appendix A.

5.7 Correlations Between Responses

Generic rules for estimating correlations between the various responses were also derived from the detailed SSMRP response

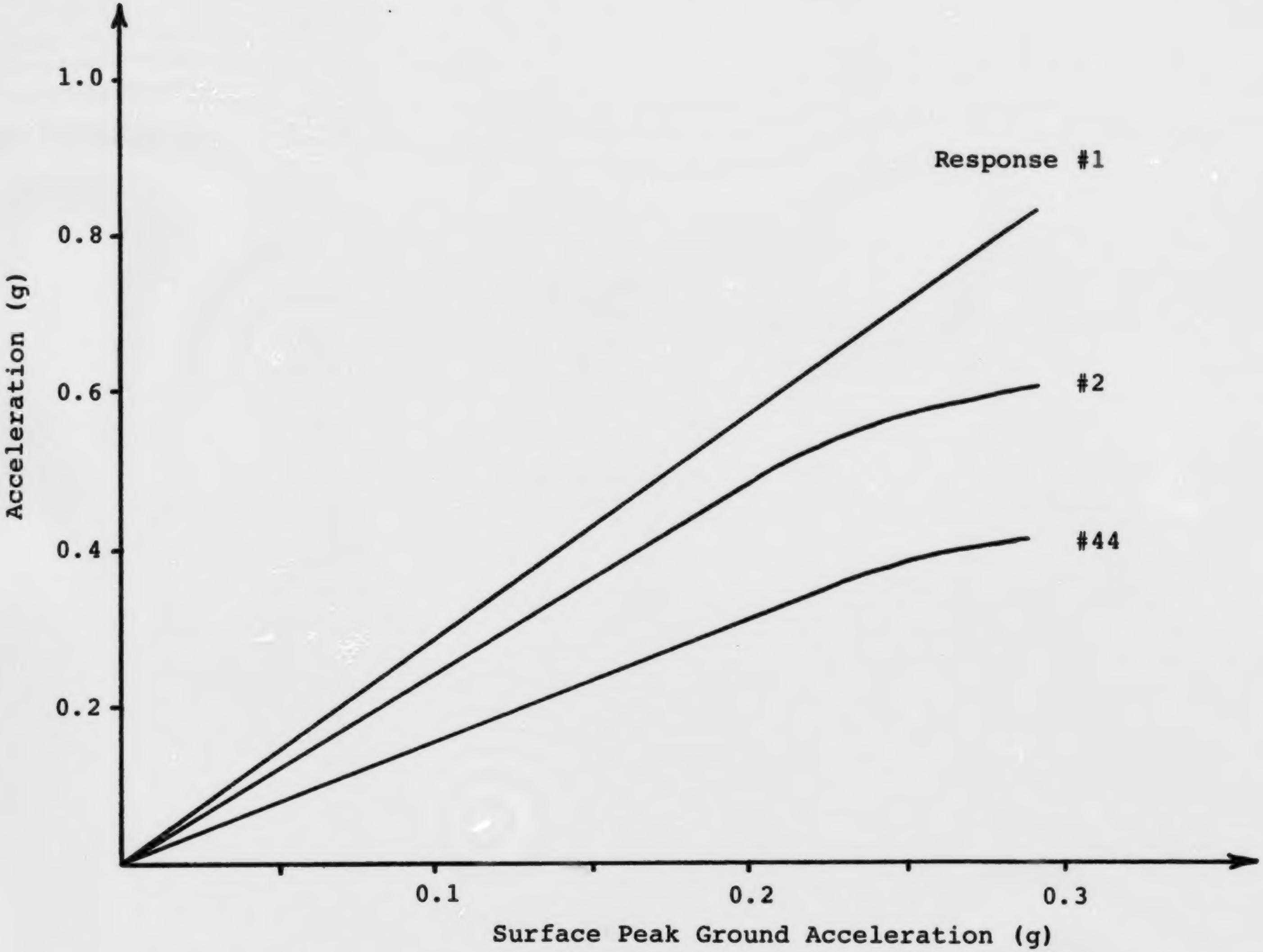

Figure 5.12 Yard and Diesel Generator Building Responses - St. Lucie

0401

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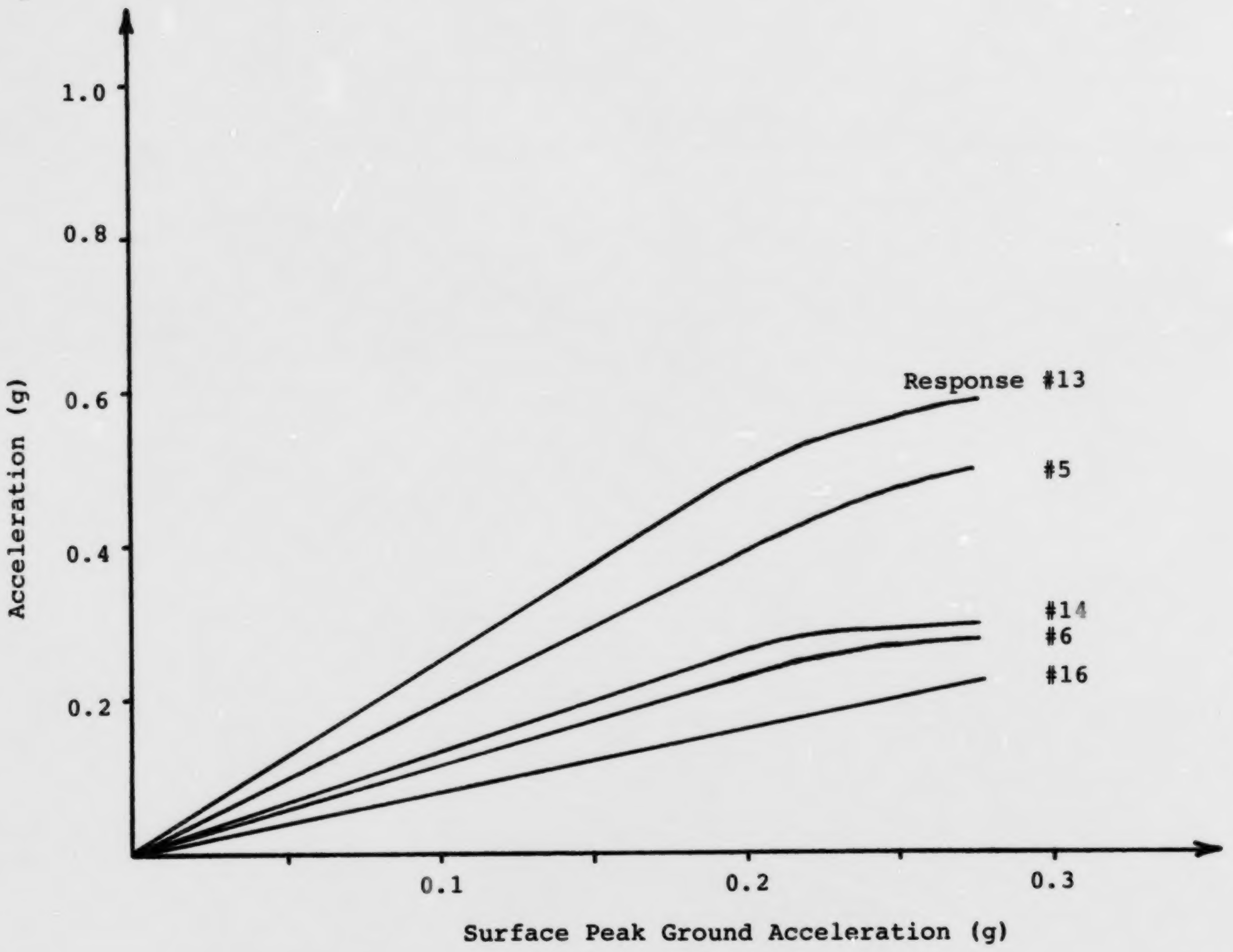

Figure 5.13 Reactor Auxiliary Building Responses - St. Lucie

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Table 5.3 St. Lucie Seismic Response Locations

Response Number	Location	Elevation †	Frequency	Multiple of Surface PGA
1	Yard	18.0	2-5	3.0
2	Yard	18.0	5-10	2.5
3	Yard	18.0	10-20	*
4	Yard	18.0	ZPA	1.0
5	RAB	-0.5	2-5	2.0
6	RAB	-0.5	5-10	1.25
7	RAB	-0.5	10-20	*
8	RAB	-0.5	ZPA	0.75
9	RAB	19.5	2-5	*
10	RAB	19.5	5-10	*
11	RAB	19.5	10-20	*
12	RAB	19.5	ZPA	*
13	RAB	43.0	2-5	2.75
14	RAB	43.0	5-10	1.50
15	RAB	43.0	10-20	*
16	RAB	43.0	ZPA	1.0
17	RAB	62.0	2-5	*
18	RAB	62.0	5-10	*
19	RAB	62.0	10-20	*
20	RAB	62.0	ZPA	*
21	RAB	82.0	2-5	*
22	RAB	82.0	5-10	2.0
23	RAB	82.0	10-20	*
24	RAB	82.0	ZPA	*
25	DGB	13.0	2-5	*
26	DGB	13.0	5-10	*
27	DGB	13.0	10-20	*
28	DGB	13.0	ZPA	*
29	DGB	23.0	2-5	*
30	DGB	23.0	5-10	*
31	DGB	23.0	10-20	*
32	DGB	23.0	ZPA	*
33	DGB	23.0	2-5	*
34	DGB	23.0	5-10	*
35	DGB	23.0	10-20	*
36	DGB	23.0	ZPA	*
37	DGB	36.0	2-5	*
38	DGB	36.0	5-10	*
39	DGB	36.0	10-20	*
40	DGB	36.0	ZPA	*
41	DGB	49.5	2-5	*
42	DGB	49.5	5-10	*
43	DGB	49.5	10-20	*
44	DGB	49.5	ZPA	1.75

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* Not used
† Elevation with respect to mean sea level
RAB = Reactor Auxiliary Building
DGB = Diesel Generator Building

0403

Table 5.4 St. Lucie Seismic Failure Basic Event Data

Basic Event	Description	Location/ Elevation	Frag. Category	Est. Freq.	Response Number	Comments
HPI1A-PMS-LF	HPI Pumps	RAB/-5	16	7	6	
HPI1B-PMS-LF		RAB/-5	16	7	6	
HPI1C-PMS-LF		RAB/-5	16	7	6	
HPI1A-BFPMS-LF	HPI Pumps	RAB/-5	16	7	6	
HPI1B-BFPMS-LF	in Bleed &	RAB/-5	16	7	6	
HPI1C-BFPMS-LF	Feed Mode	RAB/-5	16	7	6	
LPI-1A-PMS-LF	LPI Pumps	RAB/-5	16	7	6	
LPI-1B-PMS-LF		RAB/-5	16	7	6	
CSS1A-PMS-LF	Containment	RAB/-5	16	7	6	
CSS1B-PMS-LP	Spray Pumps	RAB/-5	16	7	6	
AFW1A-PMS-LF	Auxiliary	Yard/18	16	7	2	
AFW1B-PMS-LF	Feedwater	Yard/18	16	7	2	
AFW1C-PTS-LF	Pumps	Yard/18	16	7	2	
CCW1A-PMS-LF	Component	Yard/18	16	7	2	
CCW1B-PMS-LF	Cooling Water System Pumps	Yard/18	16	7	2	
ICW1A-PMS-LF	Service	Yard/18	17	5	1	
ICW1B-PMS-LF	Water	Yard/18	17	5	1	
ICW1C-PMS-LF	Pumps	Yard/18	17	5	1	

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Table 5.4 St. Lucie Seismic Failure Basic Event Data (cont.)

Basic Event	Description	Location/ Elevation	Frag. Category	Est. Freq.	Response Number	Comments
SDHX-1A-FB	RHR Heat Exchangers	RAB/-5	25	ZPA	8	
SDHX-1B-FB		RAB/-5	25	ZPA	8	
CCWHX-1A-FB	Component Cooling Water Heat Exchangers	Yard/19	23	ZPA	4	
CCWHX-1B-FB		Yard/19	23	ZPA	4	
CCW-PUMP-CM	Common Mode Pump Failure	RAB/-5	16	7	6	
HPI-PUMP-CM		RAB/-5	16	7	6	
LPI-PUMP-CM		RAB/-5	16	7	6	
CSS-PUMP-CM		RAB/-5	16	7	6	
ICW-PUMP-CM		Yard/19	17	5	1	
AFW-PUMP-CM		?	16	5	?	
CCSA-FAN-LF	Containment Fan Coolers	RB/41	28	5	13	Used RAB/13' response
CCSB-FAN-LF		RB/41	28	5	13	
CCSC-FAN-LF		RB/41	28	5	13	
CCSD-FAN-LF		RB/41	28	5	13	
RWST-A-LF	Refuelling Water Storage Tank	Yard/19	NA	NA	4	Site specific
AFW-CST1-LF	Condensate Storage Tank	Yard/19	NA	NA	4	Site specific

Table 5.4 St. Lucie Seismic Failure Basic Event Data (cont.)

Basic Event	Description	Location/ Elevation	Frag. Category	Est. Freq.	Response Number	Comments
BAT125-A-ZBT-LF	Station	RAB/43	4	ZPA	16	
BAT125-B-ZBT-LF	Batteries	RAB/43	4	ZPA	16	
BATTCHG-1A-LF	Battery	RAB/43	3	5-10	14	Used SMA value
BATTCHG-1B-LF	Chargers	RAB/43	3	5-10	14	of fragility
BATTCHG-1AA-LF		RAB/43	3	5-10	14	(Appendix B)
BATTCHG-1BB-LF		RAB/43	3	5-10	14	
DGN1-GEN-LF	Emergency	42	14	22	44	
DGN2-GEN-LF	Diesel Generators	42	14	22	44	
LOSS-OSP	Loss of Offsite Power	Yard/19	1	ZPA	4	Used ceramic insulator fragility
BUS-125-1B-LF	DC Busses	RAB/43	3	5-10	14	Used circuit
BUS-125-1AB-LF		RAB/43	3	5-10	14	breaker fragility
BUS-480-1A2-LF	480 VAC	RAB/43	3	5-10	14	Used circuit
BUS-480-1B2-LF	Busses	RAB/43	3	5-10	14	breaker fragility
BUS-416-1A2-LF	4 KV Busses	RAB/43	3	5-10	14	Used circuit
BUS-416-1B2-LF		RAB/43	3	5-10	14	breaker fragility
BUS-416-1AB-LF		RAB/43	3	5-10	14	

0406

Table 5.4 St. Lucie Seismic Failure Basic Event Data (cont.)

Basic Event	Description	Location/ Elevation	Frag. Category	Est. Freq.	Response Number	Comments
BUS-416-1A3-LF	4 KV Busses	RAB/43	3	5-10	14	Used circuit breaker fragility
BUS-416-1B3-LF		RAB/43	3	5-10	14	
MCC-480-1A5-LF	Motor Control Centers	RAB/43	3	5-10	14	Used circuit breaker fragility
MCC-480-1A6-LF		RAB/43	3	5-10	14	
MCC-480-1B5-LF		RAB/43	3	5-10	14	
MCC-480-1B6-LF		RAB/43	3	5-10	14	
CB1-800-LF	Circuit Breakers	Control Room RAB/+63	3	5-10	14	
CB2-800-LF			3	5-10	14	
CB3-800-LF			3	5-10	14	
CB4-800-LF			3	5-10	14	
SISACT-A-LF	ESF Actuation Circuit Failures	Control Room RAB/+63	3	5-10	14	Use circuit breaker fragility
SISACT-B-LF			3	5-10	14	
CASACT-A-LF			3	5-10	14	
CASACT-B-LF			3	5-10	14	
RASACT-A-LF			3	5-10	14	
RASACT-B-LF			3	5-10	14	
AFASACT-A-LF			3	5-10	14	
AFASACT-B-LF			3	5-10	14	
AFAS-SENS-CM	ESF Common Mode Sensor Failures	Inside Containment	13	5-10	22	Used maximum spectral response and el. 82' in RAB
RAS-SENS-CM					22	
CAS-SENS-CM					22	
SIS-SENS-CM					22	
BATTERY-CM			4	ZPA	16	

Table 5.4 St. Lucie Seismic Failure Basic Event Data (cont.)

Basic Event	Description	Location/ Elevation	Frag. Category	Est. Freq.	Response Number	Comments
DIESEL-CM			14	22	44	
CSS-VALVE-CM	Common Mode		18	ZPA	6*PGA	
SIS-1VALVE-CM	Valve		18	ZPA	6*PGA	
SIS-2VALVE-CM	Failures		18	ZPA	6*PGA	
SIS-2VALVE-BFCM			18	ZPA	6*PGA	
SIS-3VALVE-CM			18	ZPA	6*PGA	
SIS-3VALVE-BFCM			18	ZPA	6*PGA	
AFW-1VALVE-CM			18	ZPA	6*PGA	
AFW-2VALVE-CM			18	ZPA	6*PGA	
AFW-3VALVE-CM			18	ZPA	6*PGA	
CCW-HXVALVE-CM			18	ZPA	6*PGA	
CCW-NHVALVE-CM			18	ZPA	6*PGA	
PORV-1402-LF			20	ZPA	6*PGA	
PORV-1404-LF			20	ZPA	6*PGA	
PORV-CM			20	ZPA	6*PGA	
BLKVLV-1403-LF	Block Valves		18	ZPA	6*PGA	
BLKVLV-1405-LP			18	ZPA	6*PGA	
BLKVLV-CM			18	ZPA	6*PGA	

calculations for the Zion plant. These rules are summarized in Table 5.5. Application of these rules to the four important structures at St. Lucie results in the (symmetric) response correlation matrix given in Table 5.6. By use of this table, the correlation between any pair of responses listed in Table 5.3 can be determined. These correlations are needed in evaluating the probabilities for cut sets involving dependent failures, as described in Chapter 6.

Table 5.5

Rules for Assigning Response Correlation $\rho_{R_1R_2}$

1. Components on the same floor slab, and sensitive to the *same* spectral frequency range (i.e., ZPA, 5-10 Hz, or 10-15 Hz) will be assigned response correlation = 1.0.

2. Components on the same floor slab, sensitive to *different* ranges of spectral acceleration will be assigned response correlation = 0.5.

3. Components on different floor slabs (but in the same building) and sensitive to the *same* spectral frequency range (ZPA, 5-10 Hz or 10-15 Hz) will be assigned response correlation = 0.75.

4. Components on the ground surface (outside tanks, etc.) shall be treated as if they were on the grade floor of an adjacent building.

5. "Ganged" valve configurations (either parallel or series) will have response correlation = 1.0.

6. All other configurations will have response correlation equal to zero.

Table 5.6 Correlation Coefficients Between Responses - St. Lucie (Matrix is symmetric. Above diagonal, all blanks are zeros.)

Response Number

	1	2	4	5	6	8	13	14	16	22	44
1	1.0	0.5	0.5								
2		1.0	0.5								
4			1.0								
5				1.0	0.5	0.5	0.75				
6					1.0			0.75		0.75	
8						1.0			0.75		
		Symmetric									
13							1.0	0.5	0.5		
14								1.0	0.5	0.5	
16									1.0		
22										1.0	
44											1.0

6.0 SEISMIC FRAGILITIES

Component failure is taken as either loss of pressure boundary integrity or loss of operability. Failure (fragility) is characterized by a cumulative distribution function which describes the probability that failure has occurred given a value of loading. Loading may be described by local spectral acceleration or moment, depending on the component failure mode. The fragilities are related to the local response to permit an accurate assessment of the effect of earthquake-induced correlation in the evaluation of the accident sequences.

6.1 Generic Fragilities

A generic data base of fragility functions for seismically-induced failures was developed in the SSMRP (Reference 4). As a first step, all components were grouped into generic categories. For example, all motor operated valves located on piping with diameters between 2-1/2 and 8 inches were placed into a single generic category, and similarly, all motor control centers were placed into another generic category. The generic categories are shown on Table 6.1.

Fragility functions for the generic categories were developed based on a combination of experimental data, design analysis reports, and an extensive expert opinion survey. The experimental data utilized in developing fragility curves were obtained from the results of component manufacturer's qualification tests, independent testing lab failure data and data obtained from the extensive U.S. Corps of Engineers SAFEGUARD Subsystem Hardness Assurance Program. These data were critically examined for applicability and then statistically combined with the expert opinion survey data to produce the final fragility curves for the generic component categories (Reference 4).

Finally, since this fragility data base was assembled in the 1980-1982 time frame, a review of site-specific fragilities developed during the past four years was performed by R. Campbell of NTS Structural Mechanics Associates, Inc. (Appendix B). This review addressed the seismic fragilities developed for the PRAs of 19 different power plants. As a result of this review, the median fragility levels of 5 of the generic categories in the original SSMRP data base (Reference 4) were revised. (However, no substantial differences were found from the fragility values used in the SSMRP Zion analysis.)

Based on the above, a final set of generic fragility values was chosen for use in TAP A-45. The median failure accelerations and both random and systematic uncertainties (expressed as standard deviations of the logarithms of the fragility) are listed in Table 6.2. Values of the failure probabilities of the

Table 6-1. Fragility Categories

Fragility Category	Component Class	Typical Components	Frequency (Hz)
1	LOSP	Ceramic Insulators	ZPA
2	Relays		5-10
3	Circuit Breakers		5-10
4	Batteries		ZPA
5	Battery Racks		ZPA
6	Inverters		5-10
7	Transformers	4KV to 480V and 480 to 120V	10
8	Motor Control Centers	Control for ESP Pumps and Valves	5-10
9	Aux. Relay Cabinets		5-10
10	Switchgear (Inc. Transformers, Buses & Breakers)	4160V and 480V	5-10
11	Cable Trays		ZPA
12	Control Panels and Racks	RPS Process Control	5-10
13	Local Instruments	Misc. Pressure & Temperature Sensors	5-35
14	Diesel Generators	4160 AC Emergency Power Units	22
15	Horizontal Motors	Motor-Generator Sets	ZPA
16	Motor-Driven Pumps and Compressors	AFWS, RHR, SIS, Charging Pumps, Lube Oil Pumps, Diesel Starting Compressors	7
17	Large Vertical, Centrifugal Pumps (Motor-Drive)	Service Water Pumps	5
18	Large Motor-Operated Valves (~10")		ZPA
19	Small Motor-Operated Valves (<10")		ZPA
20	Large Pneumatic/Hydraulic Valves	Includes MSIV, ADP, and PORV	ZPA
21	Large Check and Relief Valves		ZPA
22	Miscellaneous Small Valves (<8")		ZPA
23	Large Horizontal Vessels & Heat Exchangers	Pressurizer Relief Tank, CCW Heat Exchangers	ZPA
24	Small to Medium Heat Exchangers and Vessels	Boron Injection Tank	20
25	Large Vertical Storage Vessels with Formed Heads	RHR Heat Exchanger, Accumulator Tank	ZPA
26	Large Vertical Flat-Bottomed Storage Tanks	CST, RWST	
27	Master Piping Fragility		
28	Air Handling Units	Containment Fan Coolers	5

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Table 6-2. Generic Component Fragilities, in units of Gravity (g), Except for Piping, which is in units of Moment (in.-lb.)

Category	Generic Component	Median*	β_R	β_U
1	Ceramic insulators	0.25	0.25	0.25
2	Relays	4.00	0.48	0.75
3	Circuit breakers	7.63	0.48	0.74
4	Batteries	0.80	0.40	0.39
5	Battery racks	2.29	0.31	0.39
6	Inverters	2.00	0.26	0.35
7	Dry transformers	8.80	0.28	0.30
8	Motor control centers	7.63	0.48	0.74
9	Auxiliary relay cabinets	7.63	0.48	0.74
10	Switchgear	6.43	0.29	0.66
11	Cable trays	2.23	0.34	0.19
12	Control panels and racks	11.50	0.48	0.74
13	Local instruments	7.68	0.20	0.35
14	Diesel generators	1.00	0.25	0.31
15	Horizontal motors	12.10	0.27	0.31
16	Motor-driven pumps and compressors	2.80	0.25	0.27
17	Large vertical centrifugal pumps	2.21	0.22	0.32
18	Large motor-operated valves (>10 in.)	6.50	0.26	0.60
19	Small motor-operated valves (<10 in.)	4.83	0.26	0.35
20	Large pneumatic/hydraulic valves	6.50	0.26	0.35
21	Large relief, manual, and check valves	0.90	0.20	0.35
22	Misc. small valves	12.50	0.33	0.43
23	Large horizontal vessels and heat exchangers	3.0	0.30	0.53
24	Small to medium vessels and heat exchangers	3.84	0.25	0.45
25	Large vertical vessels with formed heads	1.46	0.20	0.35
26	Large vertical tanks with flat bottoms	0.45	0.25	0.29
27	Piping (master fragility)	2.44×10^6	0.18	0.33
28	Air handling units	6.90	0.27	0.61

β_R = random uncertainty
β_U = systematic uncertainty

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*All medians in terms of spectral acceleration were developed for 5% damping.

0414

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generic component categories, as a function of peak ground acceleration level and for various floor amplification factors, are given in Appendix A. Note that these apply specifically to a site with an SSE of 0.10g. Note also that these failure probabilities have been computed based on random uncertainties in both fragility and response, and the response uncertainties were assigned as described in Section 5.5. The failure probabilities in Appendix A were used for all components except those for which site-specific fragilities were developed.

Fragilities for electrical components represent a special problem in that there is a wide variety of electrical gear found within a plant. Typically, all this gear is enclosed in switchgear or motor control centers. The two lowest failure modes in the fragility data base are relay chatter and inadvertent trip of circuit breakers. Virtually all the electrical switching and control equipment in a nuclear power plant includes these two types of components, so these two failure modes are used as generic failure modes for all electrical gear. Relay chatter is the lowest failure mode and, if included in a risk analysis, would be the dominant failure. Because, in most cases, circuits are protected by time delay circuits and because, in most cases, chatter of relays would not cause a change in the state of the system being controlled, we chose not to include relay chatter as a failure mode for electrical gear but rather included only circuit breaker trip.

6.2 Site-Specific Fragilities

A plant visit and walkdown was made in January 1986. No components were identified as requiring plant-specific fragility derivations. Two questions, however, could not be resolved:

1) Are the service water pumps of the long or short column type?

2) What were the exact dimensions of the RWST and CST vertical tanks?

Drawings for these components were never provided so as to resolve these questions. To proceed with this risk assessment, the following assumptions were made.

1) The SWS pumps were assumed to be of the long column type, which have a slightly lower median failure acceleration than the short column type (2.2g versus 2.55g). However, since these pumps did not play an important role in the results, this assumption is not of any significance.

2) The RWST and CST tanks for St. Lucie were assumed to be the same as those at the Turkey Point plant. This was based on the assumption that, since both plants were commissioned by

Florida Power and Light Company, the tanks were probably procured from and designed by the same tank vendor in the same time span. The RWST and CST did play important roles in the final risk results for St. Lucie so this assumption should be verified. The median fragilities based on the Turkey Point tanks are shown on Table 6.3.

It should be pointed out that all the components above were found to have median failure acceleration levels above the SSE. However, they did have less margin of safety above the SSE than the other components examined during the plant visit, and hence were anticipated to be significant contributors to the accident sequence probabilities.

Table 6.3 Summary of St. Lucie Site-Specific Fragility Functions

Component	Failure Mode	Median Base Acceleration at Failure	ß Random Uncertainty
RWST	Buckling with Anchor Bolt Yielding	0.24g	0.34
CST	Buckling with Anchor Bolt Yielding	0.26g	0.34

0417

7.0 CORE DAMAGE AND RISK COMPUTATIONS

7.1 Event Trees

The event trees used for the seismic analysis were taken from those used for the internal events analysis, and are shown in Figures 7.1 and 7.2. Note that failure of the reactor protection system is not in the scope of the TAP A-45 analyses, so no branches corresponding to failure to SCRAM are shown on any of the event trees.

Seven small LOCA sequences leading to core damage are identified. Loss of main feedwater (event M) occurs due to containment isolation in each sequence. The first four sequences involve cases in which the auxiliary feedwater system (L) succeeds, and thus secondary heat removal is assured, and the remaining challenge to the safety systems is to cool the core while depressurizing the primary system. This can be accomplished using the HPI system (with pressure relief via the pressurizer safety relief valves) or, if the HPI system is not available, using secondary blowdown (event X). In this latter case, rapid depressurization of the primary system is accomplished using the pressurizer sprays (and one of two CVCS pumps) in conjunction with the secondary relief (atmospheric dump) values. Following successful secondary blowdown, the low pressure system is used in both injection and recirculation phases to achieve hot shutdown conditions. Note that the RHR heat exchangers are not needed in these sequences (as denoted by the prime on event H_2) since the auxiliary feedwater system provides the necessary secondary heat removal.

Sequences 5 through 7 are cases in which the auxiliary feedwater system has failed. For these cases, a "feed and bleed" mode of cooling must be used in which coolant is injected into the primary system using the HPI system, and heat removal is accomplished by releasing steam (or water) from the pressurizer power-operated relief valves. Following the injection phase, the high pressure recirculation system (HPR) is used to continue the "feed and bleed" cooling mode.

The two transient event trees are shown in Figure 7.2. These trees are identical except that for the T_2 transient, the main feedwater system (M) is assumed not available. For event T_3, the main feedwater system is (initially) available. In each case (given that M is either not available or has failed), the auxiliary feedwater system is called upon to provide secondary cooling. Opening and closing of the pressurizer safety relief valves is required to relieve primary system overpressure. Failure of a relief valve to reclose results in a leakage equivalent to a small LOCA. If the auxiliary feedwater system fails, "feed and bleed" cooling is required to prevent core damage, as was the case for the small LOCA sequences. Since feed and bleed (event E) requires water

0418

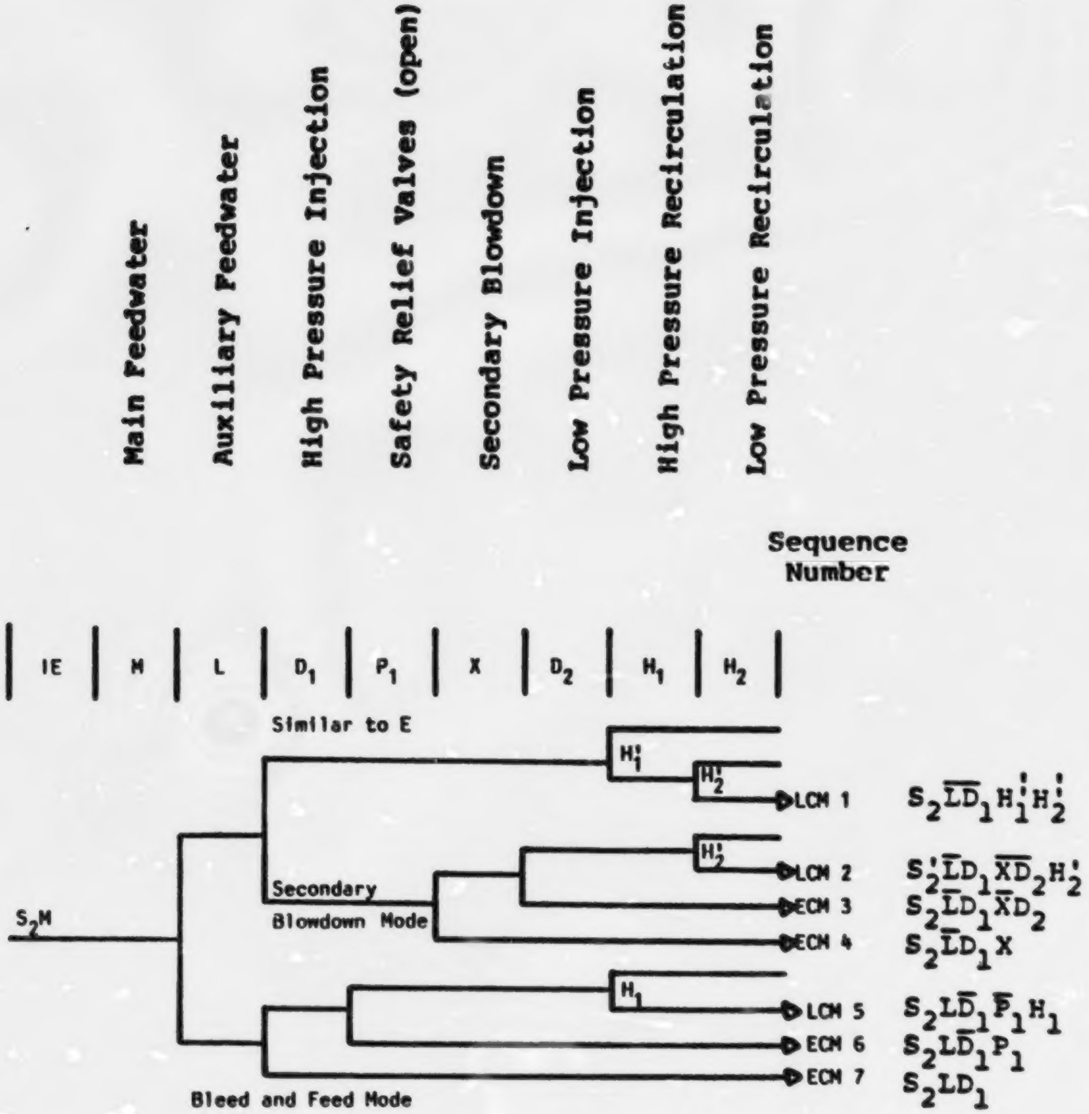
Main Feedwater
Auxiliary Feedwater
High Pressure Injection
Safety Relief Valves (open)
Secondary Blowdown
Low Pressure Injection
High Pressure Recirculation
Low Pressure Recirculation
Sequence Number
IE
M
L
D_1
P_1
X
D_2
H_1
H_2
Similar to E
H_1'
H_2'
LCM 1
$S_2\overline{L}\overline{D}_1H_1'H_2'$
H_2'
LCM 2
$S_2\overline{L}\overline{D}_1\overline{X}\overline{D}_2H_2'$
Secondary Blowdown Mode
S_2M
ECM 3
$S_2\overline{L}\overline{D}_1\overline{X}D_2$
ECM 4
$S_2\overline{L}\overline{D}_1X$
H_1
LCM 5
$S_2L\overline{D}_1\overline{P}_1H_1$
ECM 6
$S_2L\overline{D}_1P_1$
ECM 7
S_2LD_1
Bleed and Feed Mode

Figure 7-1. Small LOCA Event Tree

0419

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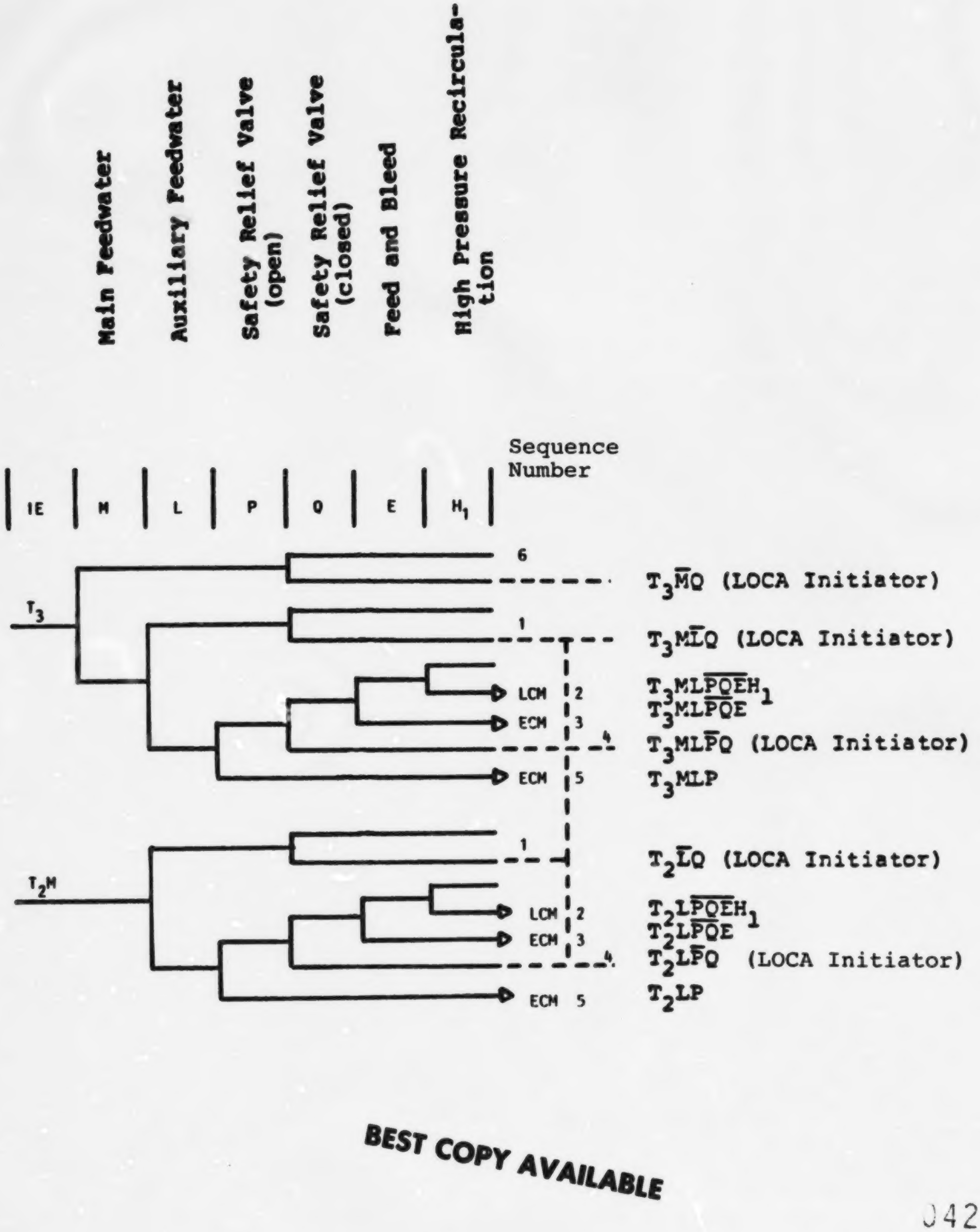
Main Feedwater
Auxiliary Feedwater
Safety Relief Valve (open)
Safety Relief Valve (closed)
Feed and Bleed
High Pressure Recirculation
Sequence Number
IE
M
L
P
Q
E
H_1
T_3
T_2M
6
$T_3\overline{M}Q$ (LOCA Initiator)
1
$T_3M\overline{L}Q$ (LOCA Initiator)
LCM 2
$T_3ML\overline{PQE}H_1$
ECM 3
$T_3ML\overline{PQ}E$
4
$T_3ML\overline{P}Q$ (LOCA Initiator)
ECM 5
T_3MLP
1
$T_2\overline{L}Q$ (LOCA Initiator)
LCM 2
$T_2L\overline{PQE}H_1$
ECM 3
$T_2L\overline{PQ}E$
4
$T_2L\overline{P}Q$ (LOCA Initiator)
ECM 5
T_2LP

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0420

from the RWST, the important accident sequences are found to be those resulting from failure of feed and bleed, in this case sequences T_2-3 and T_3-3.

7.2 Initiating Events

As discussed above, three initiating events are being considered:

(a)	Small LOCA	(pipe break with ID < 3 inches)
(b)	Transient Type 3	(PCS initially available)
(c)	Transient Type 2	(PCS initially inoperative)

The frequency of the SLOCA initiating event is taken from the calculations of pipe failure frequencies computed as part of the SSMRP. In this program, multi-support time history analysis of some 22 piping systems were made, and failure probabilities were computed for the weakest element (branch, tree, etc.) in each run of piping between valves and/or supports. These calculations included pipes from 3 inches to 29 inches and later analysis showed that the failure probabilities were not correlated with pipe size. From this data set, the probabilities of various sized LOCAs were computed, as described in Reference 3, at six levels of peak ground acceleration. The values at the four earthquake levels are shown in Table 7.1.

Table 7.1 Initiating Event Probabilities

	1 SSE	2 SSE	3 SSE	4 SSE
SLOCA	2.5E-5	8.5E-4	2.7E-2	0.09
T2	4.8E-3	0.26	0.68	0.83
T3	0.99	0.74	0.29	0.08

The frequencies of the Type 2 transient initiating event (in which loss of offsite power is implied) are computed based on the failure of ceramic insulators. These have a relatively low median failure level, and are the dominant cause of seismically-induced loss of offsite power. The probability of the T_2 transient (given that a small LOCA has not occurred) is thus

$$P(T_2) = P(\text{Ceramic Insulators})P(\overline{\text{SLOCA}})$$

Finally, the Type 3 initiating event probability is computed from the condition that the sum of the initiating event probabilities considered must be unity. The hypothesis here is that, given an earthquake of reasonable size, at least one of

0421

the initiating events will occur. Hence, we compute the probability of the Type 3 transient initiating event from:

$$P(T_3) = 1-P(T_2)-P(SLOCA)$$

The probabilities of the T_2 and T_3 initiating events at the four earthquake levels under consideration are also given in Table 7.1. At the lower earthquake levels, the probability of SLOCA is small and the transient events dominate, while the reverse is true at the higher earthquake levels.

7.3 Base Case Accident Sequence Probabilities

This section presents the results of the base case seismic risk analysis for the St. Lucie nuclear power plant. The base case is our best estimate of the current configuration of the plant and its emergency procedures. In particular, the seismic component failure probabilities were taken from the generic data base (Appendix A) with the exception of the following site-specific component failure modes:

Component	Best Estimate Failure Mode
RWST	Anchor bolt yielding and plastic buckling
CST	Anchor bolt yielding and plastic buckling

Table 7.2 lists the accident sequence probabilities at each earthquake level. These values include the initiating event probabilities (Table 7.1) and the earthquake occurrence probabilities (as given on Table 4.3). The total core melt probability (the sum of all accident sequence probabilities summed over all earthquake levels) is seen to be 1.3E-5 per year. The last column on the right hand side gives the contribution to the probability of core melt from each accident sequence. It can be seen that the dominant accident sequences are (in order of importance):

$T_2(3)$ (70%)

$T_3(3)$ (25%)

SLOCA(7) (3%)

SLOCA(3) (1%)

Taken together, these four sequences constitute 99% of the total core melt probability for the base case.

The largest contributor to the core damage is the transient sequence

$$T_2(3) = T_2\ L\bar{P}\bar{Q}E$$

Table 7.2 Base Case Accident Sequence Probabilities Per Year - St. Lucie

Sequence	Earthquake Magnitude 0.10 - 0.20g	 0.20 - 0.30g	 0.30 - 0.40g	 Above 0.40g	Total
SLOCA-1	0.00E+00	0.00E+00	0.00E+00	3.60E-12	3.60E-12
SLOCA-2	0.00E+00	0.00E+00	0.00E+00	9.60E-15	9.60E-15
SLOCA-3	3.97E-09	7.51E-08	7.73E-08	3.60E-10	1.57E-07
SLOCA-4	0.00E+00	3.65E-09	3.16E-09	9.60E-12	6.82E-09
SLOCA-5	0.00E+00	0.00E+00	8.09E-09	1.12E-13	8.09E-09
SLOCA-6	0.00E+00	1.69E-08	4.09E08	3.20E-11	5.78E-08
SLOCA-7	1.75E-09	1.02E-07	3.14E-07	1.36E-10	4.18E-07
T2-2	0.00E+00	2.63E-09	4.54E-09	6.80E-11	7.24E-09
T2-3	6.36E-07	4.59E-06	4.07E-06	6.40E-08	9.36E-06
T2-5	0.00E+00	0.00E+00	0.00E+00	2.16E-13	2.16E-13
T3-2	0.00E+00	1.54E-09	0.00E+00	4.00E-11	1.58E-09
T3-3	4.55E-07	2.20E-06	7.19E-07	4.40E-08	3.42E-06
T3-5	0.00E+00	0.00E+00	0.00E+00	1.60E-13	1.60E-13
TOTAL	1.10E-06	7.00E-06	5.24E-06	1.09E-07	1.34E-05

in which the auxiliary feedwater system (L) fails. The important cut sets in this sequence are shown in Table 7.3. This accident sequence is dominated by the two leading cut sets (failure of both the CSTs and the RWST and station blackout). The remaining cut sets are smaller by over an order of magnitude. Note that the same cut sets apply to the second most important sequence

$$T_3(3) = ML\bar{P}\bar{Q}E$$

the only difference being that, in the $T_3(3)$ sequence, loss of the main feedwater system is not given, but arises due to LOSP.

The third greatest contributor to core damage is the sequence SLOCA(7), given by

$$SLOCA(7) = S_2LD_1$$

which again involves failure of the AFWS (L) system. As shown in Table 7.4, this is again dominated by the cut sets involving failure of the two CSTs in conjunction with the RWST and station blackout. The other cut sets are at least an order of magnitude smaller.

Finally, the cut sets in sequence SLOCA(3) are listed in Table 7.5. The dominant cut sets are failure of the RWST and LOSP in conjunction with common mode failure of the diesel generators. Although this sequence is not dominant in the base case, it is the second most important sequence in the case with all seismic modifications made, as discussed in Chapter 8.0.

7.4 Release Category Assignments

Figure 7.3 shows the containment event tree used both for the internal event analysis and the seismic analysis. The tree involves success or failure of the three safety functions Y_1, C_1 and F'. Function Y_1 denotes the systems available to prevent overpressure of the containment. The functions C_1 and F' provide the post-accident radiation removal function (PARR). The functions Y_1, C_1 and F' are defined by:

$\bar{Y}_1$ = 1 out of 4 containment fan coolers

$\bar{C}_1$ = 1 out of 2 CSIS pumps

$\bar{F}'$ = 1 out of 2 CSRS pumps (without heat exchangers)

The tree defines six outcomes depending on success or failure of the containment safety systems.

Table 7.3 Dominant Cut Sets in Sequence $T_2(3)$ and $T_3(3)$
St. Lucie Base Case

	Earthquake Level				Containment
	0.1g	0.2g	0.3g	0.4g	Tree Branch
CST*RWST	1.3E-3	0.134	0.492	0.77	$\bar{Y}_1 C_1$
Battery-CM	4.4E-4	7.50E-4	6.20E-3	3.00E-2	$Y_1 C_1$
DGN1-GEN-LF*BAT125-B-ZBT-LF	6.84E-5	3.52E-4	5.44E-3	3.33E-2	$Y_1 C_1$
DGN1-GEN-LF*BAT125-B-ZBT-UTM	4.18E-5	6.05E-5	1.76E-4	3.74E-4	$Y_1 C_1$
BAT125-B-ZBT-LF*AFW1A-PMS-LF	156E-5	5.82E-5	4.76E-4	3.14E-3	$\bar{Y}_1 \bar{C}_1 \bar{F}'$
BAT125-B-ZBT-LF*AFW0909-VCC-LF	1.44E-5	5.57E-5	6.80E-4	6.08E-3	$\bar{Y}_1 \bar{C}_1 \bar{F}'$
DIESEL-CM*AFW1C-PTS-LF	1.30E-5	3.37E-5	6.02E-4	5.44E-3	$Y_1 C_1$
DIESEL-CM*AFW0803-VCC-LF	1.20E-5	3.22E-5	8.60E-4	1.05E-2	$Y_1 C_1$
DGN1-GEN-UTM*BAT125-B-ZBT-LF	1.10E-5	3.90E-5	2.07E-4	5.98E-4	$Y_1 C_1$
BAT125-B-ZBT-LF*AFASACT-A-LF	9.00E-6	3.20E-5	1.70E-4	4.90E-4	$\bar{Y}_1 \bar{C}_1 \bar{F}'$
DIESEL-CM*AFW1C-PTS-UTM	9.15E-6	2.26E-5	2.62E-4	1.04E-3	$Y_1 C_1$
BAT125-B-ZBT-LF*AFW1A-PMS-UTM	2.31E-6	1.34E-5	7.14E-5	2.06E-4	$\bar{Y}_1 \bar{C}_1 \bar{F}'$
BAT125-B-ZBT-UTM*AFW1A-PMS-LF	9.57E-6	1.00E-5	1.54E-5	3.52E-5	$\bar{Y}_1 \bar{C}_1 \bar{F}'$
BAT125-A-ZBT-LF*DGN2-GEN-LF *AFW0803-VCC-LF	ε	3.06E-6	1.09E-4	2.06E-3	$Y_1 C_1$
DIESEL-CM*AFW-2VALVE-CM	ε	1.55E-6	7.31E-5	2.21E-3	$Y_1 C_1$
DIESEL-CM*AFW-1VALVE-CM	ε	1.55E-6	7.31E-5	2.21E-3	$Y_1 C_1$
DGN2-GEN-LF*AFW0909-VCC-LF *AFW0803-VCC-LF	2.43E-6	4.16E-6	6.40E-5	1.31E-3	$\bar{Y}_1 \bar{C}_1 \bar{F}'$
DGN1-GEN-LF*AFW0803-VCC-LF *AFW0910-VCC-LF	2.43E-6	4.16E-6	6.40E-5	1.31E-3	$\bar{Y}_1 \bar{C}_1 \bar{F}'$
BAT125-B-ZBT-LF*AFW-3VALVE-CM	ε	2.69E-6	5.78E-5	1.27E-3	$\bar{Y}_1 \bar{C}_1 \bar{F}'$
BAT125-A-ZBT-LF*DGN2-GEN-LF *AFW1C-PTS-LF	ε	2.15E-6	7.62E-5	1.07E-3	$Y_1 C_1$
	1.95E-3	0.135	0.508	0.871	

0425

Table 7.4 Dominant Cut Sets in Sequence SLOCA(7)
Base Case - St. Lucie

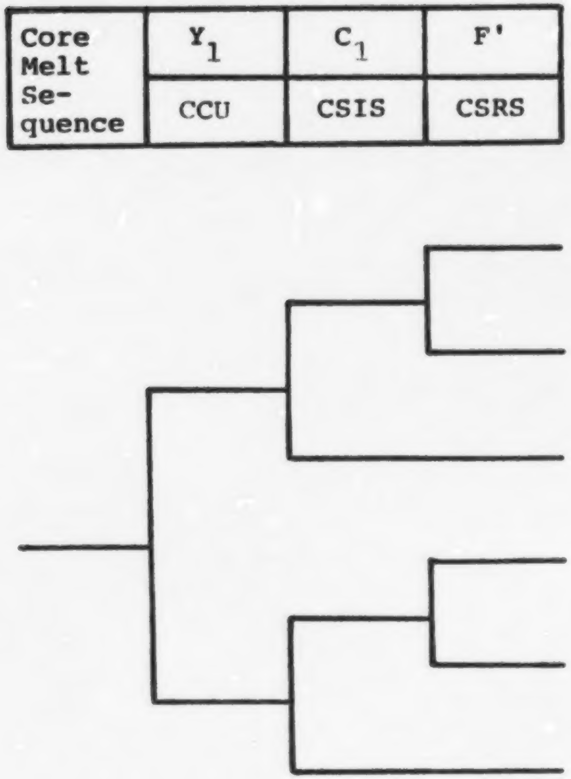
	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.3g	0.4g	
CST*RWST	1.3E-3	0.134	0.492	0.77	Y_1C_1
BATTERY-CM	4.4E-4	7.50E-4	6.20E-3	3.00E-2	Y_1C_1
DGN1-GEN-LF*BAT125-B-ZBT-LF	6.84E-5	3.52E-4	5.44E-3	3.33E-2	Y_1C_1
DGN1-GEN-LF*BAT125-B-ZBT-UTM	4.18E-5	6.05E-5	1.76E-4	3.76E-4	Y_1C_1
DIESEL-CM*AFW1C-TS-LF	1.30E-5	3.37E-5	6.02E-4	5.44E-3	Y_1C_1
DIESEL-CM*AFW0803-VCC-LF	1.20E-5	3.22E-5	8.60E-4	1.05E-2	Y_1C_1
DGN1-GEN-UTM-BAT125-B-ZBT-LF	1.10E-5	3.90E-5	2.07E-4	5.98E-4	Y_1C_1
DIESEL-CM*AFW-2VALVE-CM	ϵ	1.55E-6	7.31E-5	2.21E-3	Y_1C_1
DIESEL-CM*AFW-1VALVE-CM	ϵ	1.55E-6	7.31E-5	2.21E-3	Y_1C_1
BAT125-A-ZBT-LF*DGN2-GEN-LF *AFW0803-VCC-LF	ϵ	3.06E-6	1.09E-4	2.06E-3	Y_1C_1
BAT125-A-ZBT-LF*DGN2-GEN-LF *AFW1C-PTS-LF	ϵ	3.20E-6	7.62E-5	1.07E-3	Y_1C_1
DIESEL-CM*AFW1C-PTS-UTM	9.15E-6	2.26E-5	2.62E-4	1.04E-3	Y_1C_1

0426

Table 7.5 Dominant Cut Sets in Sequence SLOCA(3)
Base Case - St. Lucie

	Earthquake Level				Containment Tree Branch
	0.1g	0.2g	0.3g	0.4g	
RWST	1.9E-2	0.33	0.70	0.88	
DIESEL-CM	1.5E-3	3.7E-3	4.3E-2	0.17	Y_1C_1
CCW1B-PMS-LF*DGN1-GEN-LF	3.3E-4	5.0E-4	2.2E-3	1.1E-2	Y_1C_1
CCW1A-PMS-LF*DGN2-GEN-LF	3.3E-4	5.0E-4	2.2E-3	1.1E-2	Y_1C_1
ICW1A-PMS-LF*DGN2-GEN-LF	3.3E-4	7.1E-4	7.2E-3	4.1E-2	Y_1C_1
DGN1-GEN-LF*DGN2-GEN-UTM	2.3E-4	3.4E-4	9.8E-4	2.1E-3	Y_1C_1
DGN1-GEN-UTM*DGN2-GEN-LF	2.3E-4	3.4E-4	9.8E-4	2.1E-3	Y_1C_1
SISACT-B-LF*DGN1-GEN-LF	1.9E-4	2.8E-4	8.0E-4	1.7E-3	Y_1C_1
SISACT-A-LF*DGN2-GEN-LF	1.9E-4	2.8E-4	8.0E-4	1.7E-3	Y_1C_1
CW1A-PMS-LF*CCW1B-PMS-LF	7.6E-5	8.3E-5	2.0E-4	1.0E-3	Y_1C_1
CCW1B-PMS-LF*ICW1A-PMS-LF	7.6E-5	1.2E-4	6.3E-4	3.8E-3	Y_1C_1
BAT125-A-ZBT-LF*DGN2-GEN-LF	6.8E-5	3.5E-4	5.4E-3	3.3E-2	Y_1C_1
ICW-PUMP-CM	1.0E-5	3.2E-4	6.8E-3	3.6E-2	Y_1C_1
DGN1-GEN-LF*SIS071B-VOC-LF	1.4E-6	4.1E-5	1.9E-3	1.8E-2	$\bar{Y}_1C_1$
DGN2-GEN-LF*SIS071A-VOC-LF	1.4E-6	4.1E-5	1.9E-3	1.8E-2	$\bar{Y}_1C_1$
ICW144B-VOC-LF*DGN1-GEN-LF	1.4E-6	4.1E-5	1.9E-3	1.8E-2	Y_1C_1
ICW144A-VOC-LF*DGN2-GEN-LF	1.4E-6	4.1E-5	1.9E-3	1.8E-2	Y_1C_1

0427



Status of Containment Overpressure Protection	Status of Post Accident Radioactivity Removal	Containment Systems Sequence
Success	Success	$\bar{Y}_1\bar{C}_1\bar{F}'$
Success	Late Failure	$\bar{Y}_1\bar{C}_1F'$
Success	Early Failure	$\bar{Y}_1C_1$
Success	Success	$Y_1\bar{C}_1\bar{F}'$
Success	Late Failure	$Y_1\bar{C}_1F'$
Failure	Early Failure	Y_1C_1

Figure 7.3 Containment Systems Event Tree

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The mapping from the six containment tree sequences to radioactive release categories was based on MARCH code studies performed as part of the Calvert Cliffs RSSMAP study (Reference 17). This plant has a containment size and systems design similar to that at St. Lucie. The mapping, as used both for the internal event analysis and the seismic analysis, is given on Table 7.6.

In order to apportion the accident sequence core damage probabilities to the appropriate branch of the containment event tree, the most general approach is to determine the cut sets associated with each containment tree branch, and combine these (using Boolean logic) with the accident sequence probabilities. This general approach is not necessary in this case, however, due to the fact that the core damage probabilities were dominated by (1) cut sets involving failure of both CSTs and the RWST, (2) electrical component failure cut sets, and (3) failure of the RWST alone.

The assignments of the cutsets in each dominant accident sequence to the containment tree branches are listed in Tables 7.3, 7.4, and 7.5. The fractional contribution of each cutset to the accident sequence probability is computed in Table 7.7.

With these assignments, and the mapping of the containment event tree branches to release categories given in Table 7.6, one can compute the annual probability of release in each category as shown in Table 7.8. Finally, the population dose (in man-REM/year) is estimated using the source term per release category values, taken as 0.3 times the WASH-1400 values, as shown below:

Release Category	Annual Probability	Source Term	Dose (man-REM/yr)
1	1.3E-9	8.5E+5	0.001
2	1.9E-7	8.6E+5	0.163
3	2.3E-6	6.7E+5	1.54
4	2.6E-8	3.9E+5	0.010
5	2.3E-8	2.0E+5	0.005
6	3.3E-6	3.2E+4	0.106
7	3.3E-8	4.8E+3	ε
		Total man-REM/yr =	1.83

Thus, for the base case, a total of 1.83 man-REM/year is estimated due to the seismic hazard.

Table 7.6 St. Lucie Accident Sequence to Release Category Mapping

Containment Systems Sequence*	Special Conditions	Containment Failure Mode with Probability and Release Category				
	Core Melt	α	β	γ, δ_e	δ_ℓ	ϵ
$\bar{Y}_1\bar{C}_1\bar{F}'$		1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 5	2.5E-1 7
$\bar{Y}_1\bar{C}_1F'$	EM	1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 4	2.5E-1 6
	LM	1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6
$\bar{Y}_1C_1$		1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6
$Y_1\bar{C}_1\bar{F}'$		1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 5	2.5E-1 7
$Y_1\bar{C}_1F'$	EM	1E-4 1	2E-3 5	1.4E-2 3	1.8E-1 4	2.5E-1 6
	LM	1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6
Y_1C_1		1E-4 1	2E-3 4	1.4E-2 2	1.8E-1 3	2.5E-1 6

*The exact definition of the containment system sequence varies for the PWRs depending upon the system success criteria.

EM ≡ early core melt, LM ≡ late core melt.

0430

Table 7.7 Partition of Accident Sequence Frequencies to Containment Branches, St. Lucie Base Case

		0.1-0.2g	0.2-0.3g	0.3-0.4g	>0.4g	Average Fraction
$T_2L\bar{P}\bar{Q}E$	P(SEQ)	6.4E-7	4.6E-6	4.1E-6	6.4E-8	
	$\bar{Y}_1C_1$	67%	99%	97%	88%	96%
	Y_1C	30%	0.95%	2.8%	10%	3%
	$\bar{Y}_1\bar{C}\bar{F}'$	3%	0.05%	0.2%	2%	1%
$T_3ML\bar{P}\bar{Q}E$		4.6E-7	2.2E-6	7.2E-7	4.4E-8	
	$\bar{Y}_1C_1$	67%	99%	97%	88%	94%
	Y_1C	30%	0.95%	2.8%	10%	5%
	$\bar{Y}_1\bar{C}\bar{F}'$	3%	0.05%	0.2%	2%	1%
S_2LD_1		1.8E-9	1.0E-7	3.1E-7	1.4E-10	
	$\bar{Y}_1C_1$	71%	99%	97%	89%	96%
	Y_1C_1	29%	1%	3%	11%	4%

0431

Table 7.8 Calculation of Annual Release Category Probabilities, St. Lucie Base Case

Sequence (Probability)	Average Fraction		Release Categories						
			1	2	3	4	5	6	7
$T_2L\bar{P}\bar{Q}E$	96%	$\bar{Y}_1C_1$(1)	9.0E-10	1.3E-7	1.6E-6	1.8E-8	ε	2.3E-6	ε
(9.4E-6)	3%	Y_1C_1	2.8E-11	4.0E-9	5.1E-8	5.6E-10	ε	7.1E-8	ε
	1%	$\bar{Y}_1\bar{C}_1\bar{F}'$	9.4E-12	ε	1.3E-9	ε	1.7E-8	ε	2.4E-8
$T_3ML\bar{P}\bar{Q}E$	94%	$\bar{Y}_1C_1$(1)	3.2E-10	4.5E-8	5.8E-7	6.4E-9	ε	8.0E-7	ε
(3.4E-6)	5%	Y_1C_1	1.7E-11	2.4E-9	3.1E-8	3.4E-10	ε	4.3E-8	ε
	1%	$\bar{Y}_1\bar{C}_1\bar{F}'$	3.4E-12	ε	4.8E-10	ε	6.2E-9	ε	8.5E-9
S_2LD_1	96%	$\bar{Y}_1C_1$(1)	4.0E-11	5.6E-9	7.3E-8	8.1E-10	ε	1.0E-7	ε
(4.2E-7)	4%	Y_1C_1	1.7E-12	2.4E-10	3.0E-9	3.4E-11	ε	4.2E-9	ε
		TOTAL	1.3E-9	1.9E-7	2.3E-6	2.6E-8	2.3E-8	3.3E-6	3.3E-8

(1) Due to CST * RWST cutset only.

8.0 POTENTIAL DESIGN CHANGES AND RISK IMPACT

Based on the plant walk-through and the base case evaluation of the risk of seismically-induced core melt at the St. Lucie plant, a number of potential modifications were identified, and the potential reduction in risk resulting from these modifications was evaluated as discussed below.

8.1 Potential Modifications

Both the CSTs and the RWSTs are located in the yard (elevation 18 feet). Based on the TAP A-45 seismic analysis of Turkey Point, both are assumed to have low seismic resistance, with the failure mode of shell buckling near the base and anchor bolt yielding. The proposed modification to these tanks would be to add structural stiffening so as to increase the median ground acceleration level at failure to approximately 4 times the SSE which would significantly reduce their contribution to the probability of core damage.

Table 8.1 presents the recalculated core melt probabilities for the varying earthquake levels if all of these modifications were implemented. (In these calculations, it is assumed that the probabilities of failure of the aforementioned items are reduced to zero.) The total core melt probability with these modifications incorporated would be reduced to approximately 9.6E-7 per year.

Table 8.2 presents the assignment of sequences to the containment event tree branches. Table 8.3 shows the mapping of the core damage sequence annual probabilities, using the mapping given in Table 7.5. Finally, the population dose (in man-REM/year) is again estimated using the source term per release category values, taken as 0.3 times the WASH-1400 values, as shown below:

Release Category	Annual Probability	Source Term	Dose (man-REM/yr)
1	9.3E-11	8.5E+5	ε
2	1.3E-8	8.6E+5	0.01
3	1.6E-7	6.7E+5	0.11
4	1.8E-9	3.9E+5	ε
5	8.2E-9	2.0E+5	ε
6	1.5E-7	3.2E+4	0.01
7	4.0E-8	4.8E+3	ε
		Total	0.13

Thus, for the case when all modifications are made, the estimated annual population dose due to earthquakes alone is 0.13 man-REM/yr, which is an order of magnitude decrease relative to the base case.

Table 8.1 Accident Sequence Probabilities per Year - All Modifications Made - St. Lucie

	Earthquake Magnitude				
Sequence	**0.1 - 0.2g**	**0.2 - 0.3g**	**0.3 - 0.4g**	**Above 0.4g**	**Total**
SLOCA-1	0.00E+00	2.26E-09	1.53E-08	6.80E-12	1.76E-08
SLOCA-2	0.00E+00	0.00E+00	0.00E+00	1.60E-14	1.60E-14
SLOCA-3	0.00E+00	1.50E-08	1.24E-07	2.16E-11	1.39E-07
SLOCA-4	0.00E+00	0.00E+00	6.29E-09	8.00E-13	6.29E-09
SLOCa-5	0.00E+00	0.00E+00	0.00E+00	2.72E-15	2.72E-15
SLOCA-6	0.00E+00	0.00E+00	3.00E-09	2.96E-13	3.00E-09
SLOCA-7	0.00E+00	2.83E-09	4.65E-08	2.72E-12	4.93E-08
T2-2	0.00E+00	0.00E+00	0.00E+00	7.20E-13	7.20E-13
T2-3	1.01E-08	1.09E-07	5.09E-07	9.60E-10	6.29E-07
T2-5	0.00E+00	0.00E+00	0.00E+00	1.68E-15	1.68E-15
T3-2	0.00E+00	0.00E-00	0.00E-00	5.92E-13	5.92E-13
T3-3	9.24E-09	4.56E-08	6.11E-08	9.60E-10	1.17E-07
T3-5	0.00E+00	0.00E+00	0.00E+00	1.44E-15	1.44E-15
Total	1.93E-08	1.75E-07	7.65E-07	1.95E-09	9.61E-07

0434

Table 8.2 Partition of Accident Sequence Frequencies to Containment Tree Branches, St. Lucie Case With All Modifications Made

Dominant Sequence	Dominant Cut Sets	Cut Set Fractions at Each Earthquake Level				Average Fraction
		0.1-0.2g	0.2-0.3g	0.3-0.4g	> 70.4g	
$T_2L\bar{P}\bar{Q}E$	P(SEQ)	1.0E-8	1.1E-7	5.1E-7	9.6E-10	
	Y_1C_1	91%	95%	93%	83%	94%
	$\bar{Y}_1\bar{C}_1\bar{F}'$	9%	5%	7%	17%	6%
$T_3ML\bar{P}\bar{Q}E$		9.2E-9	4.6E-8	6.1E-8	9.6E-10	
	Y_1C_1	91%	95%	93%	83%	94%
	$\bar{Y}_1\bar{C}_1\bar{F}'$	9%	5%	7%	17%	6%
$S_2\bar{L}D_1\bar{X}D_2$		ε	1.5E-8	1.2E-7	2.2E-11	
	Y_1C_1	100%	100%	100%	100%	100%
S_2LD_1		ε	2.8E-9	4.7E-8	2.7E-12	
	Y_1C_1	100%	100%	100%	100%	100%

0435

Table 8.3 Calculation of Annual Release Category Probabilities, St. Lucie - Case With All Modifications Made

Dominant Sequence (Probability)	Average Fraction		Release Categories						
			1	2	3	4	5	6	7
$T_2L\bar{P}\bar{Q}E$	94%	Y_1C_1	5.9E-11	8.3E-9	1.1E-7	1.2E-9	ε	1.5E-7	ε
(6.3E-7)	6%	$\bar{Y}_1\bar{C}_1\bar{F}'$	3.8E-12	ε	5.3E-10	ε	6.9E-9	ε	3.8E-8
$T_3ML\bar{P}\bar{Q}E$	94%	Y_1C_1	1.1E-11	1.6E-9	2.0E-8	2.3E-10	ε	2.8E-8	ε
(1.2E-7)	6%	$\bar{Y}_1\bar{C}_1\bar{F}'$	7.2E-13	ε	1.0E-10	ε	1.3E-9	ε	1.8E-9
$S_2\bar{L}D_1\bar{X}D_2$ (1.4E-7)	100%	Y_1C_1	1.4E-11	2.0E-9	2.5E-8	2.8E-10	ε	3.5E-8	ε
S_2LD_1 (4.9E-8)	100%	Y_1C_1	4.9E-12	6.9E-10	8.8E-9	9.8E-11	ε	1.2E-8	ε
		TOTAL	9.3E-11	1.3E-8	1.64E-7	1.81E-9	8.2E-9	1.5E-7	4.0E-8

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9.0 SUMMARY AND CONCLUSIONS

Two cases of the seismic risk at St. Lucie have been analyzed, a base case which includes a number of potentially vulnerable site-specific items judged to have relatively small margin of safety over the SSE, and a case with all significant vulnerable components upgraded to the point where their contribution to the seismic risk is negligible. In both cases, credit was taken for the feed and bleed cooling and secondary blowdown options.

The base case has the largest probability of core damage (1.3E-5 per year) and an estimated exposure of 1.8 man-REM/year. The base case is dominated (95%) by two transient sequences involving failure of the AFWS. The dominant contributor to these sequences is common mode failure of the two CSTs and the RWST. Station blackout (LOSP plus common mode failure of the station batteries) is the next most important contributor. Only one LOCA sequence (SLOCA-7) made any non-negligible contribution to the base case, and its overall contribution was very small (3% of core damage frequency) due to the small probability of the SLOCA initiating event (see Table 7.1).

In the second case analyzed, the CSTs and RWST are assumed to be upgraded so that their seismic contributions are negligible. In this case, the frequency of core damage drops to 9.6E-7 per year, and the exposure drops to 0.13 man-REM/year. The dominant sequence (66%) is again $T_2(3)$, with the remaining sequences $T_3(3)$ and $S_2(3)$ each contributing approximately 15% to the total core damage frequency. Blackout cut sets dominate the transient sequences, while LOSP and battery or diesel generator failures dominate the $S_2(3)$ sequence.

10.0 REFERENCES

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APPENDIX C.1

Generic Fragilities for SSE = 0.10g

PAGE = 1

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY				
					1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
CERAMIC INSULATORS	0.25	0.25	0.25	0.50	2.7E-06	4.8E-03	7.4E-02	2.6E-01	5.0E-01
				1.00	4.8E-03	2.6E-01	7.0E-01	9.1E-01	9.8E-01
				1.25	2.5E-02	5.0E-01	8.7E-01	9.8E-01	1.0E+00
				1.50	7.4E-02	7.0E-01	9.5E-01	9.9E-01	1.0E+00
				1.75	1.6E-01	8.3E-01	9.8E-01	1.0E+00	1.0E+00
				2.00	2.6E-01	9.1E-01	9.9E-01	1.0E+00	1.0E+00
				2.25	3.8E-01	9.5E-01	1.0E+00	1.0E+00	1.0E+00
				2.50	5.0E-01	9.8E-01	1.0E+00	1.0E+00	1.0E+00
				2.75	6.1E-01	9.9E-01	1.0E+00	1.0E+00	1.0E+00
				3.00	7.0E-01	9.9E-01	1.0E+00	1.0E+00	1.0E+00
				3.25	7.7E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				3.50	8.3E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				3.75	8.7E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.00	9.1E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.25	9.3E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.50	9.5E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				4.75	9.7E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.00	9.8E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.25	9.8E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.50	9.9E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				5.75	9.9E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
				6.00	9.9E-01	1.0E+00	1.0E+00	1.0E+00	1.0E+00
RELAYS	4.00	0.48	0.45	0.50	1.4E-11	1.0E-08	3.0E-07	2.6E-06	1.3E-05
				1.00	1.0E-08	2.6E-06	4.1E-05	2.3E-04	7.9E-04
				1.25	6.9E-08	1.3E-05	1.6E-04	7.9E-04	2.4E-03
				1.50	3.0E-07	4.1E-05	4.5E-04	2.0E-03	5.5E-03
				1.75	9.9E-07	1.1E-04	1.0E-03	4.0E-03	1.0E-02
				2.00	2.6E-06	2.3E-04	2.0E-03	7.2E-03	1.8E-02
				2.25	6.1E-06	4.5E-04	3.4E-03	1.2E-02	2.7E-02
				2.50	1.3E-05	7.9E-04	5.5E-03	1.8E-02	3.9E-02
				2.75	2.4E-05	1.3E-03	8.2E-03	2.5E-02	5.2E-02
				3.00	4.1E-05	2.0E-03	1.2E-02	3.4E-02	6.8E-02
				3.25	6.8E-05	2.9E-03	1.6E-02	4.4E-02	8.5E-02
				3.50	1.1E-04	4.0E-03	2.1E-02	5.5E-02	1.0E-01
				3.75	1.6E-04	5.5E-03	2.7E-02	6.8E-02	1.2E-01
				4.00	2.3E-04	7.2E-03	3.4E-02	8.2E-02	1.5E-01
				4.25	3.3E-04	9.3E-03	4.1E-02	9.7E-02	1.7E-01
				4.50	4.5E-04	1.2E-02	4.9E-02	1.1E-01	1.9E-01
				4.75	6.0E-04	1.4E-02	5.8E-02	1.3E-01	2.1E-01
				5.00	7.9E-04	1.8E-02	6.8E-02	1.5E-01	2.4E-01
				5.25	1.0E-03	2.1E-02	7.8E-02	1.6E-01	2.6E-01
				5.50	1.3E-03	2.5E-02	8.9E-02	1.8E-01	2.8E-01
				5.75	1.6E-03	2.9E-02	1.0E-01	2.0E-01	3.1E-01
				6.00	2.0E-03	3.4E-02	1.1E-01	2.2E-01	3.3E-01

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PAGE = 2

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
CIRCUIT BREAKERS	7.63	0.48	0.45	0.50	1.1E-14	2.2E-11	1.2E-09	1.6E-08	1.0E-07
				1.00	2.2E-11	1.6E-08	4.4E-07	3.7E-06	1.7E-05
				1.25	2.1E-10	1.0E-07	2.3E-06	1.7E-05	7.2E-05
				1.50	1.2E-09	4.4E-07	8.5E-06	5.6E-05	2.1E-04
				1.75	4.8E-09	1.4E-06	2.4E-05	1.4E-04	5.0E-04
				2.00	1.6E-08	3.7E-06	5.6E-05	3.0E-04	1.0E-03
				2.25	4.3E-08	8.5E-06	1.1E-04	5.8E-04	1.8E-03
				2.50	1.0E-07	1.7E-05	2.1E-04	1.0E-03	3.0E-03
				2.75	2.2E-07	3.2E-05	3.6E-04	1.6E-03	4.6E-03
				3.00	4.4E-07	5.6E-05	5.8E-04	2.5E-03	6.7E-03
				3.25	8.1E-07	9.1E-05	8.8E-04	3.6E-03	9.4E-03
				3.50	1.4E-06	1.4E-04	1.3E-03	5.0E-03	1.3E-02
				3.75	2.3E-06	2.1E-04	1.8E-03	6.7E-03	1.6E-02
				4.00	3.7E-06	3.0E-04	2.5E-03	8.8E-03	2.1E-02
				4.25	5.7E-06	4.3E-04	3.3E-03	1.1E-02	2.6E-02
				4.50	8.5E-06	5.8E-04	4.2E-03	1.4E-02	3.2E-02
				4.75	1.2E-05	7.7E-04	5.4E-03	1.7E-02	3.8E-02
				5.00	1.7E-05	1.0E-03	6.7E-03	2.1E-02	4.5E-02
				5.25	2.4E-05	1.3E-03	8.2E-03	2.5E-02	5.2E-02
				5.50	3.2E-05	1.6E-03	1.0E-02	2.9E-02	6.0E-02
				5.75	4.3E-05	2.0E-03	1.2E-02	3.4E-02	6.9E-02
				6.00	5.6E-05	2.5E-03	1.4E-02	3.9E-02	7.8E-02
BATTERIES	0.80	0.40	0.35	0.50	9.1E-08	4.6E-05	8.2E-04	4.6E-03	1.4E-02
				1.00	4.6E-05	4.6E-03	3.2E-02	9.6E-02	1.9E-01
				1.25	2.4E-04	1.4E-02	7.7E-02	1.9E-01	3.2E-01
				1.50	8.2E-04	3.2E-02	1.4E-01	2.9E-01	4.5E-01
				1.75	2.1E-03	6.0E-02	2.1E-01	4.0E-01	5.7E-01
				2.00	4.6E-03	9.6E-02	2.9E-01	5.0E-01	6.6E-01
				2.25	8.5E-03	1.4E-01	3.7E-01	5.9E-01	7.4E-01
				2.50	1.4E-02	1.9E-01	4.5E-01	6.6E-01	8.0E-01
				2.75	2.2E-02	2.4E-01	5.2E-01	7.3E-01	8.5E-01
				3.00	3.2E-02	2.9E-01	5.9E-01	7.8E-01	8.8E-01
				3.25	4.5E-02	3.5E-01	6.5E-01	8.2E-01	9.1E-01
				3.50	6.0E-02	4.0E-01	7.0E-01	8.5E-01	9.3E-01
				3.75	7.7E-02	4.5E-01	7.4E-01	8.8E-01	9.5E-01
				4.00	9.6E-02	5.0E-01	7.8E-01	9.0E-01	9.6E-01
				4.25	1.2E-01	5.5E-01	8.1E-01	9.2E-01	9.7E-01
				4.50	1.4E-01	5.9E-01	8.4E-01	9.4E-01	9.7E-01
				4.75	1.6E-01	6.3E-01	8.6E-01	9.5E-01	9.8E-01
				5.00	1.9E-01	6.6E-01	8.8E-01	9.6E-01	9.8E-01
				5.25	2.1E-01	7.0E-01	9.0E-01	9.7E-01	9.9E-01
				5.50	2.4E-01	7.3E-01	9.1E-01	9.7E-01	9.9E-01
				5.75	2.7E-01	7.5E-01	9.3E-01	9.8E-01	9.9E-01
				6.00	2.9E-01	7.8E-01	9.4E-01	9.8E-01	9.9E-01

BEST COPY AVAILABLE

PAGE = 3

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
BATTERY RACKS	2.29	0.31	0.35	0.50	1.4E-16	1.1E-11	2.8E-09	9.2E-08	1.1E-06
				1.00	1.1E-11	9.2E-08	6.9E-06	9.5E-05	5.7E-04
				1.25	2.5E-10	1.1E-06	5.4E-05	5.7E-04	2.7E-03
				1.50	2.8E-09	6.9E-06	2.5E-04	2.1E-03	8.5E-03
				1.75	1.9E-08	2.9E-05	8.2E-04	5.6E-03	2.0E-02
				2.00	9.2E-08	9.5E-05	2.1E-03	1.2E-02	3.8E-02
				2.25	3.5E-07	2.5E-04	4.5E-03	2.3E-02	6.4E-02
				2.50	1.1E-06	5.7E-04	8.5E-03	3.8E-02	9.8E-02
				2.75	2.9E-06	1.1E-03	1.4E-02	5.8E-02	1.4E-01
				3.00	6.9E-06	2.1E-03	2.3E-02	8.3E-02	1.8E-01
				3.25	1.5E-05	3.5E-03	3.4E-02	1.1E-01	2.3E-01
				3.50	2.9E-05	5.6E-03	4.8E-02	1.5E-01	2.8E-01
				3.75	5.4E-05	8.5E-03	6.4E-02	1.8E-01	3.3E-01
				4.00	9.5E-05	1.2E-02	8.3E-02	2.2E-01	3.9E-01
				4.25	1.6E-04	1.7E-02	1.1E-01	2.6E-01	4.4E-01
				4.50	2.5E-04	2.3E-02	1.3E-01	3.0E-01	4.8E-01
				4.75	3.8E-04	3.0E-02	1.6E-01	3.4E-01	5.3E-01
				5.00	5.7E-04	3.8E-02	1.8E-01	3.9E-01	5.7E-01
				5.25	8.2E-04	4.8E-02	2.1E-01	4.3E-01	6.1E-01
				5.50	1.1E-03	5.8E-02	2.4E-01	4.7E-01	6.5E-01
				5.75	1.6E-03	7.0E-02	2.7E-01	5.0E-01	6.9E-01
				6.00	2.1E-03	8.3E-02	3.0E-01	5.4E-01	7.2E-01
INVERTERS	2.00	0.26	0.45	0.50	6.3E-13	4.1E-09	3.1E-07	4.7E-06	3.2E-05
				1.00	4.1E-09	4.7E-06	1.3E-04	9.8E-04	3.8E-03
				1.25	4.8E-08	3.2E-05	6.4E-04	3.8E-03	1.3E-02
				1.50	3.1E-07	1.3E-04	2.1E-03	1.0E-02	3.0E-02
				1.75	1.4E-06	4.0E-04	5.0E-03	2.2E-02	5.6E-02
				2.00	4.7E-06	9.8E-04	1.0E-02	3.9E-02	9.1E-02
				2.25	1.3E-05	2.1E-03	1.8E-02	6.2E-02	1.3E-01
				2.50	3.2E-05	3.8E-03	3.0E-02	9.1E-02	1.8E-01
				2.75	6.7E-05	6.5E-03	4.4E-02	1.3E-01	2.4E-01
				3.00	1.3E-04	1.0E-02	6.2E-02	1.6E-01	2.9E-01
				3.25	2.4E-04	1.5E-02	8.3E-02	2.0E-01	3.4E-01
				3.50	4.0E-04	2.2E-02	1.1E-01	2.5E-01	4.0E-01
				3.75	6.4E-04	3.0E-02	1.3E-01	2.9E-01	4.5E-01
				4.00	9.8E-04	3.9E-02	1.6E-01	3.3E-01	5.0E-01
				4.25	1.4E-03	5.0E-02	1.9E-01	3.8E-01	5.5E-01
				4.50	2.1E-03	6.2E-02	2.2E-01	4.2E-01	5.9E-01
				4.75	2.8E-03	7.6E-02	2.6E-01	4.6E-01	6.3E-01
				5.00	3.8E-03	9.1E-02	2.9E-01	5.0E-01	6.7E-01
				5.25	5.0E-03	1.1E-01	3.2E-01	5.4E-01	7.0E-01
				5.50	6.5E-03	1.3E-01	3.6E-01	5.7E-01	7.3E-01
				5.75	8.2E-03	1.4E-01	3.9E-01	6.1E-01	7.6E-01
				6.00	1.0E-02	1.6E-01	4.2E-01	6.4E-01	7.8E-01

BEST COPY AVAILABLE

PAGE = 4

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY				
					1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
TRANSFORMERS	8.80	0.28	0.45	0.50	8.7E-23	1.5E-17	7.8E-15	4.7E-13	9.2E-12
				1.00	1.5E-17	4.7E-13	9.2E-11	2.7E-09	3.1E-08
				1.25	5.0E-16	9.2E-12	1.3E-09	3.1E-08	3.0E-07
				1.50	7.8E-15	9.2E-11	1.0E-08	2.0E-07	1.7E-06
				1.75	7.2E-14	5.9E-10	5.2E-08	8.9E-07	6.6E-06
				2.00	4.7E-13	2.7E-09	2.0E-07	3.0E-06	2.0E-05
				2.25	2.3E-12	1.0E-08	6.3E-07	8.5E-06	5.2E-05
				2.50	9.2E-12	3.1E-08	1.7E-06	2.0E-05	1.2E-04
				2.75	3.1E-11	8.4E-08	4.0E-06	4.4E-05	2.3E-04
				3.00	9.2E-11	2.0E-07	8.5E-06	8.5E-05	4.2E-04
				3.25	2.4E-10	4.4E-07	1.7E-05	1.5E-04	7.2E-04
				3.50	5.9E-10	8.9E-07	3.0E-05	2.6E-04	1.2E-03
				3.75	1.3E-09	1.7E-06	5.2E-05	4.2E-04	1.8E-03
				4.00	2.7E-09	3.0E-06	8.5E-05	6.5E-04	2.6E-03
				4.25	5.4E-09	5.2E-06	1.3E-04	9.6E-04	3.7E-03
				4.50	1.0E-08	8.5E-06	2.0E-04	1.4E-03	5.0E-03
				4.75	1.8E-08	1.3E-05	3.0E-04	1.9E-03	6.7E-03
				5.00	3.1E-08	2.0E-05	4.2E-04	2.6E-03	8.8E-03
				5.25	5.2E-08	3.0E-05	5.8E-04	3.4E-03	1.1E-02
				5.50	8.4E-08	4.4E-05	7.9E-04	4.5E-03	1.4E-02
				5.75	1.3E-07	6.2E-05	1.1E-03	5.7E-03	1.7E-02
				6.00	2.0E-07	8.5E-05	1.4E-03	7.1E-03	2.1E-02
MOTOR CONTRL CENTERS-MCC	7.63	0.48	0.45	0.50	1.1E-14	2.2E-11	1.2E-09	1.6E-08	1.0E-07
				1.00	2.2E-11	1.6E-08	4.4E-07	3.7E-06	1.7E-05
				1.25	2.1E-10	1.0E-07	2.3E-06	1.7E-05	7.2E-05
				1.50	1.2E-09	4.4E-07	8.5E-06	5.6E-05	2.1E-04
				1.75	4.8E-09	1.4E-06	2.4E-05	1.4E-04	5.0E-04
				2.00	1.6E-08	3.7E-06	5.6E-05	3.0E-04	1.0E-03
				2.25	4.3E-08	8.5E-06	1.1E-04	5.8E-04	1.8E-03
				2.50	1.0E-07	1.7E-05	2.1E-04	1.0E-03	3.0E-03
				2.75	2.2E-07	3.2E-05	3.6E-04	1.6E-03	4.6E-03
				3.00	4.4E-07	5.6E-05	5.8E-04	2.5E-03	6.7E-03
				3.25	8.1E-07	9.1E-05	8.8E-04	3.6E-03	9.4E-03
				3.50	1.4E-06	1.4E-04	1.3E-03	5.0E-03	1.3E-02
				3.75	2.3E-06	2.1E-04	1.8E-03	6.7E-03	1.6E-02
				4.00	3.7E-06	3.0E-04	2.5E-03	8.8E-03	2.1E-02
				4.25	5.7E-06	4.3E-04	3.3E-03	1.1E-02	2.6E-02
				4.50	8.5E-06	5.8E-04	4.2E-03	1.4E-02	3.2E-02
				4.75	1.2E-05	7.7E-04	5.4E-03	1.7E-02	3.8E-02
				5.00	1.7E-05	1.0E-03	6.7E-03	2.1E-02	4.5E-02
				5.25	2.4E-05	1.3E-03	8.2E-03	2.5E-02	5.2E-02
				5.50	3.2E-05	1.6E-03	1.0E-02	2.9E-02	6.0E-02
				5.75	4.3E-05	2.0E-03	1.2E-02	3.4E-02	6.9E-02
				6.00	5.6E-05	2.5E-03	1.4E-02	3.9E-02	7.8E-02

BEST COPY AVAILABLE

PAGE = 5

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
AUX RELAY CABINETS	7.63	0.48	0.45	0.50	1.1E-14	2.2E-11	1.2E-09	1.6E-08	1.0E-07
				1.00	2.2E-11	1.6E-08	4.4E-07	3.7E-06	1.7E-05
				1.25	2.1E-10	1.0E-07	2.3E-06	1.7E-05	7.2E-05
				1.50	1.2E-09	4.4E-07	8.5E-06	5.6E-05	2.1E-04
				1.75	4.8E-09	1.4E-06	2.4E-05	1.4E-04	5.0E-04
				2.00	1.6E-08	3.7E-06	5.6E-05	3.0E-04	1.0E-03
				2.25	4.3E-08	8.5E-06	1.1E-04	5.8E-04	1.8E-03
				2.50	1.0E-07	1.7E-05	2.1E-04	1.0E-03	3.0E-03
				2.75	2.2E-07	3.2E-05	3.6E-04	1.6E-03	4.6E-03
				3.00	4.4E-07	5.6E-05	5.8E-04	2.5E-03	6.7E-03
				3.25	8.1E-07	9.1E-05	8.8E-04	3.6E-03	9.4E-03
				3.50	1.4E-06	1.4E-04	1.3E-03	5.0E-03	1.3E-02
				3.75	2.3E-06	2.1E-04	1.8E-03	6.7E-03	1.6E-02
				4.00	3.7E-06	3.0E-04	2.5E-03	8.8E-03	2.1E-02
				4.25	5.7E-06	4.3E-04	3.3E-03	1.1E-02	2.6E-02
				4.50	8.5E-06	5.8E-04	4.2E-03	1.4E-02	3.2E-02
				4.75	1.2E-05	7.7E-04	5.4E-03	1.7E-02	3.8E-02
				5.00	1.7E-05	1.0E-03	6.7E-03	2.1E-02	4.5E-02
				5.25	2.4E-05	1.3E-03	8.2E-03	2.5E-02	5.2E-02
				5.50	3.2E-05	1.6E-03	1.0E-02	2.9E-02	6.0E-02
				5.75	4.3E-05	2.0E-03	1.2E-02	3.4E-02	6.9E-02
				6.00	5.6E-05	2.5E-03	1.4E-02	3.9E-02	7.8E-02
SWITCHGEAR	6.43	0.29	0.45	0.50	5.8E-20	3.7E-15	1.1E-12	4.5E-11	6.6E-10
				1.00	3.7E-15	4.5E-11	5.2E-09	1.1E-07	9.2E-07
				1.25	9.2E-14	6.6E-10	5.5E-08	9.2E-07	6.7E-06
				1.50	1.1E-12	5.2E-09	3.4E-07	4.7E-06	3.0E-05
				1.75	8.4E-12	2.7E-08	1.4E-06	1.7E-05	9.7E-05
				2.00	4.5E-11	1.1E-07	4.7E-06	5.0E-05	2.5E-04
				2.25	1.9E-10	3.4E-07	1.3E-05	1.2E-04	5.6E-04
				2.50	6.6E-10	9.2E-07	3.0E-05	2.5E-04	1.1E-03
				2.75	2.0E-09	2.2E-06	6.3E-05	4.9E-04	2.0E-03
				3.00	5.2E-09	4.7E-06	1.2E-04	8.6E-04	3.3E-03
				3.25	1.2E-08	9.3E-06	2.1E-04	1.4E-03	5.1E-03
				3.50	2.7E-08	1.7E-05	3.6E-04	2.2E-03	7.5E-03
				3.75	5.5E-08	3.0E-05	5.6E-04	3.3E-03	1.1E-02
				4.00	1.1E-07	5.0E-05	8.6E-04	4.7E-03	1.5E-02
				4.25	1.9E-07	7.8E-05	1.3E-03	6.5E-03	1.9E-02
				4.50	3.4E-07	1.2E-04	1.8E-03	8.7E-03	2.5E-02
				4.75	5.7E-07	1.8E-04	2.4E-03	1.1E-02	3.1E-02
				5.00	9.2E-07	2.5E-04	3.3E-03	1.5E-02	3.9E-02
				5.25	1.4E-06	3.6E-04	4.3E-03	1.8E-02	4.7E-02
				5.50	2.2E-06	4.9E-04	5.5E-03	2.3E-02	5.6E-02
				5.75	3.2E-06	6.5E-04	7.0E-03	2.7E-02	6.6E-02
				6.00	4.7E-06	8.6E-04	8.7E-03	3.3E-02	7.7E-02

BEST COPY AVAILABLE

PAGE = 6

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY				
					1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
CABLE TRAYS	2.23	0.34	0.35	0.50	3.5E-15	9.9E-11	1.6E-08	3.9E-07	3.7E-06
				1.00	9.9E-11	3.9E-07	2.0E-05	2.1E-04	1.1E-03
				1.25	1.8E-09	3.7E-06	1.3E-04	1.1E-03	4.6E-03
				1.50	1.6E-08	2.0E-05	5.2E-04	3.6E-03	1.3E-02
				1.75	9.2E-08	7.4E-05	1.5E-03	8.8E-03	2.8E-02
				2.00	3.9E-07	2.1E-04	3.6E-03	1.8E-02	5.0E-02
				2.25	1.3E-06	5.2E-04	7.2E-03	3.1E-02	8.0E-02
				2.50	3.7E-06	1.1E-03	1.3E-02	5.0E-02	1.2E-01
				2.75	9.0E-06	2.1E-03	2.1E-02	7.4E-02	1.6E-01
				3.00	2.0E-05	3.6E-03	3.1E-02	1.0E-01	2.1E-01
				3.25	4.0E-05	5.8E-03	4.5E-02	1.3E-01	2.6E-01
				3.50	7.4E-05	8.8E-03	6.1E-02	1.7E-01	3.1E-01
				3.75	1.3E-04	1.3E-02	8.0E-02	2.1E-01	3.6E-01
				4.00	2.1E-04	1.8E-02	1.0E-01	2.5E-01	4.1E-01
				4.25	3.4E-04	2.4E-02	1.3E-01	2.9E-01	4.6E-01
				4.50	5.2E-04	3.1E-02	1.5E-01	3.3E-01	5.1E-01
				4.75	7.6E-04	4.0E-02	1.8E-01	3.7E-01	5.5E-01
				5.00	1.1E-03	5.0E-02	2.1E-01	4.1E-01	5.9E-01
				5.25	1.5E-03	6.1E-02	2.4E-01	4.5E-01	6.3E-01
				5.50	2.1E-03	7.4E-02	2.7E-01	4.9E-01	6.7E-01
				5.75	2.7E-03	8.7E-02	3.0E-01	5.3E-01	7.0E-01
				6.00	3.6E-03	1.0E-01	3.3E-01	5.6E-01	7.3E-01
CONTRL PANEL & RACKS	11.50	0.48	0.45	0.50	7.0E-17	2.8E-13	2.1E-11	3.7E-10	3.0E-09
				1.00	2.8E-13	3.7E-10	1.5E-08	1.7E-07	9.4E-07
				1.25	3.2E-12	3.0E-09	9.8E-08	9.4E-07	4.8E-06
				1.50	2.1E-11	1.5E-08	4.2E-07	3.6E-06	1.7E-05
				1.75	1.0E-10	5.6E-08	1.4E-06	1.0E-05	4.5E-05
				2.00	3.7E-10	1.7E-07	3.6E-06	2.5E-05	1.0E-04
				2.25	1.1E-09	4.2E-07	8.2E-06	5.4E-05	2.1E-04
				2.50	3.0E-09	9.4E-07	1.7E-05	1.0E-04	3.7E-04
				2.75	7.0E-09	1.9E-06	3.1E-05	1.8E-04	6.2E-04
				3.00	1.5E-08	3.6E-06	5.4E-05	3.0E-04	9.8E-04
				3.25	3.0E-08	6.3E-06	8.8E-05	4.6E-04	1.5E-03
				3.50	5.6E-08	1.0E-05	1.4E-04	6.9E-04	2.1E-03
				3.75	9.8E-08	1.7E-05	2.1E-04	9.8E-04	2.9E-03
				4.00	1.7E-07	2.5E-05	3.0E-04	1.4E-03	3.9E-03
				4.25	2.7E-07	3.8E-05	4.1E-04	1.8E-03	5.1E-03
				4.50	4.2E-07	5.4E-05	5.7E-04	2.4E-03	6.6E-03
				4.75	6.4E-07	7.5E-05	7.5E-04	3.1E-03	8.3E-03
				5.00	9.4E-07	1.0E-04	9.8E-04	3.9E-03	1.0E-02
				5.25	1.4E-06	1.4E-04	1.3E-03	4.9E-03	1.2E-02
				5.50	1.9E-06	1.8E-04	1.6E-03	6.0E-03	1.5E-02
				5.75	2.6E-06	2.3E-04	2.0E-03	7.2E-03	1.8E-02
				6.00	3.6E-06	3.0E-04	2.4E-03	8.6E-03	2.1E-02

BEST COPY AVAILABLE

PAGE = 7

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY				
					1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
LOCAL INSTRUMENTS	7.68	0.20	0.45	0.50	7.8E-25	5.9E-19	6.6E-16	6.4E-14	1.8E-12
				1.00	5.9E-19	6.4E-14	2.3E-11	9.8E-10	1.4E-08
				1.25	3.1E-17	1.8E-12	4.4E-10	1.4E-08	1.8E-07
				1.50	6.6E-16	2.3E-11	4.2E-09	1.1E-07	1.2E-06
				1.75	8.0E-15	1.8E-10	2.5E-08	5.7E-07	5.1E-06
				2.00	6.4E-14	9.8E-10	1.1E-07	2.2E-06	1.7E-05
				2.25	3.8E-13	4.2E-09	3.9E-07	6.7E-06	4.8E-05
				2.50	1.8E-12	1.4E-08	1.2E-06	1.7E-05	1.1E-04
				2.75	6.8E-12	4.3E-08	2.9E-06	4.0E-05	2.4E-04
				3.00	2.3E-11	1.1E-07	6.7E-06	8.2E-05	4.6E-04
				3.25	6.7E-11	2.7E-07	1.4E-05	1.5E-04	8.1E-04
				3.50	1.8E-10	5.7E-07	2.7E-05	2.7E-04	1.3E-03
				3.75	4.4E-10	1.2E-06	4.8E-05	4.6E-04	2.1E-03
				4.00	9.8E-10	2.2E-06	8.2E-05	7.2E-04	3.1E-03
				4.25	2.1E-09	3.9E-06	1.3E-04	1.1E-03	4.5E-03
				4.50	4.2E-09	6.7E-06	2.1E-04	1.6E-03	6.3E-03
				4.75	8.0E-09	1.1E-05	3.1E-04	2.3E-03	8.6E-03
				5.00	1.4E-08	1.7E-05	4.6E-04	3.1E-03	1.1E-02
				5.25	2.5E-08	2.7E-05	6.5E-04	4.2E-03	1.5E-02
				5.50	4.3E-08	4.0E-05	9.0E-04	5.6E-03	1.9E-02
				5.75	7.1E-08	5.8E-05	1.2E-03	7.2E-03	2.3E-02
				6.00	1.1E-07	8.2E-05	1.6E-03	9.1E-03	2.8E-02
DIESEL GENERATORS	1.00	0.25	0.35	0.50	1.6E-12	4.3E-08	5.2E-06	9.1E-05	6.3E-04
				1.00	4.3E-08	9.1E-05	2.6E-03	1.7E-02	5.4E-02
				1.25	6.7E-07	6.3E-04	1.1E-02	5.4E-02	1.4E-01
				1.50	5.2E-06	2.6E-03	3.2E-02	1.2E-01	2.5E-01
				1.75	2.5E-05	7.3E-03	6.7E-02	2.0E-01	3.8E-01
				2.00	9.1E-05	1.7E-02	1.2E-01	3.0E-01	5.0E-01
				2.25	2.6E-04	3.2E-02	1.8E-01	4.0E-01	6.1E-01
				2.50	6.3E-04	5.4E-02	2.5E-01	5.0E-01	7.0E-01
				2.75	1.3E-03	8.2E-02	3.3E-01	5.9E-01	7.7E-01
				3.00	2.6E-03	1.2E-01	4.0E-01	6.6E-01	8.3E-01
				3.25	4.5E-03	1.6E-01	4.8E-01	7.3E-01	8.7E-01
				3.50	7.3E-03	2.0E-01	5.5E-01	7.8E-01	9.0E-01
				3.75	1.1E-02	2.5E-01	6.1E-01	8.3E-01	9.3E-01
				4.00	1.7E-02	3.0E-01	6.6E-01	8.6E-01	9.5E-01
				4.25	2.3E-02	3.5E-01	7.1E-01	8.9E-01	9.6E-01
				4.50	3.2E-02	4.0E-01	7.6E-01	9.1E-01	9.7E-01
				4.75	4.2E-02	4.5E-01	7.9E-01	9.3E-01	9.8E-01
				5.00	5.4E-02	5.0E-01	8.3E-01	9.5E-01	9.8E-01
				5.25	6.7E-02	5.5E-01	8.5E-01	9.6E-01	9.9E-01
				5.50	8.2E-02	5.9E-01	8.8E-01	9.7E-01	9.9E-01
				5.75	9.9E-02	6.3E-01	9.0E-01	9.7E-01	9.9E-01
				6.00	1.2E-01	6.6E-01	9.1E-01	9.8E-01	9.9E-01

BEST COPY AVAILABLE

PAGE = 8

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
HORIZONTAL MOTORS	12.10	0.27	0.35	0.50	1.1E-35	1.0E-27	1.5E-23	8.4E-21	8.4E-19
				1.00	1.0E-27	8.4E-21	3.0E-17	6.1E-15	2.8E-13
				1.25	2.2E-25	8.4E-19	1.9E-15	2.8E-13	1.0E-11
				1.50	1.5E-23	3.0E-17	4.8E-14	5.4E-12	1.6E-10
				1.75	4.7E-22	5.5E-16	6.3E-13	5.7E-11	1.4E-09
				2.00	8.4E-21	6.1E-15	5.4E-12	4.0E-10	8.5E-09
				2.25	9.9E-20	4.8E-14	3.3E-11	2.1E-09	3.9E-08
				2.50	8.4E-19	2.8E-13	1.6E-10	8.5E-09	1.4E-07
				2.75	5.6E-18	1.3E-12	6.2E-10	2.9E-08	4.3E-07
				3.00	3.0E-17	5.4E-12	2.1E-09	8.6E-08	1.2E-06
				3.25	1.4E-16	1.9E-11	6.1E-09	2.2E-07	2.8E-06
				3.50	5.5E-16	5.7E-11	1.6E-08	5.3E-07	6.1E-06
				3.75	1.9E-15	1.6E-10	3.9E-08	1.2E-06	1.2E-05
				4.00	6.1E-15	4.0E-10	8.6E-08	2.4E-06	2.3E-05
				4.25	1.8E-14	9.4E-10	1.8E-07	4.5E-06	4.2E-05
				4.50	4.8E-14	2.1E-09	3.5E-07	8.1E-06	7.1E-05
				4.75	1.2E-13	4.3E-09	6.5E-07	1.4E-05	1.2E-04
				5.00	2.8E-13	8.5E-09	1.2E-06	2.3E-05	1.8E-04
				5.25	6.3E-13	1.6E-08	2.0E-06	3.7E-05	2.7E-04
				5.50	1.3E-12	2.9E-08	3.3E-06	5.8E-05	4.0E-04
				5.75	2.7E-12	5.1E-08	5.2E-06	8.6E-05	5.7E-04
				6.00	5.4E-12	8.6E-08	8.1E-06	1.3E-04	8.0E-04
MOTOR-DRIVEN PUMPS/COMPR	2.80	0.25	0.45	0.50	2.7E-15	4.8E-11	6.5E-09	1.5E-07	1.3E-06
				1.00	4.8E-11	1.5E-07	7.2E-06	7.8E-05	4.1E-04
				1.25	7.7E-10	1.3E-06	4.7E-05	4.1E-04	1.8E-03
				1.50	6.5E-09	7.2E-06	1.9E-04	1.4E-03	5.2E-03
				1.75	3.6E-08	2.7E-05	5.7E-04	3.5E-03	1.2E-02
				2.00	1.5E-07	7.8E-05	1.4E-03	7.5E-03	2.3E-02
				2.25	4.8E-07	1.9E-04	2.9E-03	1.4E-02	3.8E-02
				2.50	1.3E-06	4.1E-04	5.2E-03	2.3E-02	5.9E-02
				2.75	3.3E-06	7.8E-04	8.8E-03	3.5E-02	8.4E-02
				3.00	7.2E-06	1.4E-03	1.4E-02	5.0E-02	1.1E-01
				3.25	1.4E-05	2.3E-03	2.0E-02	6.8E-02	1.5E-01
				3.50	2.7E-05	3.5E-03	2.8E-02	8.9E-02	1.8E-01
				3.75	4.7E-05	5.2E-03	3.8E-02	1.1E-01	2.2E-01
				4.00	7.8E-05	7.5E-03	5.0E-02	1.4E-01	2.6E-01
				4.25	1.2E-04	1.0E-02	6.3E-02	1.7E-01	3.0E-01
				4.50	1.9E-04	1.4E-02	7.8E-02	2.0E-01	3.4E-01
				4.75	2.8E-04	1.8E-02	9.5E-02	2.3E-01	3.7E-01
				5.00	4.1E-04	2.3E-02	1.1E-01	2.6E-01	4.1E-01
				5.25	5.7E-04	2.8E-02	1.3E-01	2.9E-01	4.5E-01
				5.50	7.8E-04	3.5E-02	1.5E-01	3.2E-01	4.9E-01
				5.75	1.1E-03	4.2E-02	1.7E-01	3.5E-01	5.2E-01
				6.00	1.4E-03	5.0E-02	2.0E-01	3.8E-01	5.5E-01

BEST COPY AVAILABLE

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
LG VERT M-D CENTRIF PUMP	2.21	0.22	0.45	0.50	2.0E-14	3.2E-10	3.9E-08	8.1E-07	6.8E-06
				1.00	3.2E-10	8.1E-07	3.3E-05	3.2E-04	1.5E-03
				1.25	4.9E-09	6.8E-06	2.0E-04	1.5E-03	5.8E-03
				1.50	3.9E-08	3.3E-05	7.4E-04	4.6E-03	1.5E-02
				1.75	2.1E-07	1.2E-04	2.1E-03	1.1E-02	3.2E-02
				2.00	8.1E-07	3.2E-04	4.6E-03	2.1E-02	5.7E-02
				2.25	2.5E-06	7.4E-04	8.9E-03	3.6E-02	8.9E-02
				2.50	6.8E-06	1.5E-03	1.5E-02	5.7E-02	1.3E-01
				2.75	1.6E-05	2.7E-03	2.5E-02	8.2E-02	1.7E-01
				3.00	3.3E-05	4.6E-03	3.6E-02	1.1E-01	2.2E-01
				3.25	6.5E-05	7.3E-03	5.1E-02	1.4E-01	2.7E-01
				3.50	1.2E-04	1.1E-02	6.9E-02	1.8E-01	3.2E-01
				3.75	2.0E-04	1.5E-02	8.9E-02	2.2E-01	3.7E-01
				4.00	3.2E-04	2.1E-02	1.1E-01	2.6E-01	4.2E-01
				4.25	5.0E-04	2.8E-02	1.4E-01	3.0E-01	4.7E-01
				4.50	7.4E-04	3.6E-02	1.6E-01	3.4E-01	5.1E-01
				4.75	1.1E-03	4.6E-02	1.9E-01	3.8E-01	5.6E-01
				5.00	1.5E-03	5.7E-02	2.2E-01	4.2E-01	6.0E-01
				5.25	2.1E-03	6.9E-02	2.5E-01	4.6E-01	6.3E-01
				5.50	2.7E-03	8.2E-02	2.8E-01	5.0E-01	6.7E-01
				5.75	3.6E-03	9.6E-02	3.1E-01	5.3E-01	7.0E-01
				6.00	4.6E-03	1.1E-01	3.4E-01	5.7E-01	7.3E-01
LARGE MOV	6.50	0.26	0.35	0.50	3.1E-29	5.1E-22	2.7E-18	7.1E-16	3.9E-14
				1.00	5.1E-22	7.1E-16	8.7E-13	8.0E-11	2.0E-09
				1.25	6.4E-20	3.9E-14	3.0E-11	2.0E-09	3.9E-08
				1.50	2.7E-18	8.7E-13	4.5E-10	2.3E-08	3.7E-07
				1.75	5.6E-17	1.0E-11	3.9E-09	1.6E-07	2.1E-06
				2.00	7.1E-16	8.0E-11	2.3E-08	7.7E-07	8.8E-06
				2.25	6.1E-15	4.5E-10	1.0E-07	2.9E-06	2.9E-05
				2.50	3.9E-14	2.0E-09	3.7E-07	8.8E-06	7.8E-05
				2.75	2.0E-13	7.4E-09	1.1E-06	2.3E-05	1.8E-04
				3.00	8.7E-13	2.3E-08	2.9E-06	5.3E-05	3.9E-04
				3.25	3.2E-12	6.4E-08	6.8E-06	1.1E-04	7.4E-04
				3.50	1.0E-11	1.6E-07	1.5E-05	2.1E-04	1.3E-03
				3.75	3.0E-11	3.7E-07	2.9E-05	3.9E-04	2.2E-03
				4.00	8.0E-11	7.7E-07	5.3E-05	6.5E-04	3.4E-03
				4.25	2.0E-10	1.5E-06	9.4E-05	1.0E-03	5.2E-03
				4.50	4.5E-10	2.9E-06	1.6E-04	1.6E-03	7.5E-03
				4.75	9.8E-10	5.2E-06	2.5E-04	2.4E-03	1.0E-02
				5.00	2.0E-09	8.8E-06	3.9E-04	3.4E-03	1.4E-02
				5.25	3.9E-09	1.5E-05	5.7E-04	4.8E-03	1.9E-02
				5.50	7.4E-09	2.3E-05	8.3E-04	6.5E-03	2.4E-02
				5.75	1.3E-08	3.6E-05	1.2E-03	8.6E-03	3.1E-02
				6.00	2.3E-08	5.3E-05	1.6E-03	1.1E-02	3.8E-02

BEST COPY AVAILABLE

PAGE =10

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY				
					1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
SMALL MOV SMALL AOV	4.83	0.26	0.35	0.50	5.2E-26	3.0E-19	8.4E-16	1.4E-13	5.5E-12
				1.00	3.0E-19	1.4E-13	9.2E-11	5.5E-09	9.9E-08
				1.25	2.6E-17	5.5E-12	2.3E-09	9.9E-08	1.4E-06
				1.50	8.4E-16	9.2E-11	2.6E-08	8.6E-07	9.7E-06
				1.75	1.4E-14	8.7E-10	1.8E-07	4.7E-06	4.5E-05
				2.00	1.4E-13	5.5E-09	8.6E-07	1.9E-05	1.5E-04
				2.25	1.0E-12	2.6E-08	3.2E-06	5.8E-05	4.2E-04
				2.50	5.5E-12	9.9E-08	9.7E-06	1.5E-04	9.7E-04
				2.75	2.5E-11	3.1E-07	2.5E-05	3.5E-04	2.0E-03
				3.00	9.2E-11	8.6E-07	5.8E-05	7.0E-04	3.7E-03
				3.25	3.0E-10	2.1E-06	1.2E-04	1.3E-03	6.2E-03
				3.50	8.7E-10	4.7E-06	2.3E-04	2.3E-03	9.9E-03
				3.75	2.3E-09	9.7E-06	4.2E-04	3.7E-03	1.5E-02
				4.00	5.5E-09	1.9E-05	7.0E-04	5.6E-03	2.2E-02
				4.25	1.2E-08	3.4E-05	1.1E-03	8.3E-03	3.0E-02
				4.50	2.6E-08	5.8E-05	1.7E-03	1.2E-02	4.0E-02
				4.75	5.2E-08	9.6E-05	2.6E-03	1.6E-02	5.2E-02
				5.00	9.9E-08	1.5E-04	3.7E-03	2.2E-02	6.5E-02
				5.25	1.8E-07	2.3E-04	5.1E-03	2.8E-02	8.1E-02
				5.50	3.1E-07	3.5E-04	6.9E-03	3.6E-02	9.8E-02
				5.75	5.3E-07	5.0E-04	9.1E-03	4.4E-02	1.2E-01
				6.00	8.6E-07	7.0E-04	1.2E-02	5.4E-02	1.4E-01
LG PNEUM/HYD VALVE	6.50	0.26	0.35	0.50	3.1E-29	5.1E-22	2.7E-18	7.1E-16	3.9E-14
				1.00	5.1E-22	7.1E-16	8.7E-13	8.0E-11	2.0E-09
				1.25	6.4E-20	3.9E-14	3.0E-11	2.0E-09	3.9E-08
				1.50	2.7E-18	8.7E-13	4.5E-10	2.3E-08	3.7E-07
				1.75	5.6E-17	1.0E-11	3.9E-09	1.6E-07	2.1E-06
				2.00	7.1E-16	8.0E-11	2.3E-08	7.7E-07	8.8E-06
				2.25	6.1E-15	4.5E-10	1.0E-07	2.9E-06	2.9E-05
				2.50	3.9E-14	2.0E-09	3.7E-07	8.8E-06	7.8E-05
				2.75	2.0E-13	7.4E-09	1.1E-06	2.3E-05	1.8E-04
				3.00	8.7E-13	2.3E-08	2.9E-06	5.3E-05	3.9E-04
				3.25	3.2E-12	6.4E-08	6.8E-06	1.1E-04	7.4E-04
				3.50	1.0E-11	1.6E-07	1.5E-05	2.1E-04	1.3E-03
				3.75	3.0E-11	3.7E-07	2.9E-05	3.9E-04	2.2E-03
				4.00	8.0E-11	7.7E-07	5.3E-05	6.5E-04	3.4E-03
				4.25	2.0E-10	1.5E-06	9.4E-05	1.0E-03	5.2E-03
				4.50	4.5E-10	2.9E-06	1.6E-04	1.6E-03	7.5E-03
				4.75	9.8E-10	5.2E-06	2.5E-04	2.4E-03	1.0E-02
				5.00	2.0E-09	8.8E-06	3.9E-04	3.4E-03	1.4E-02
				5.25	3.9E-09	1.5E-05	5.7E-04	4.8E-03	1.9E-02
				5.50	7.4E-09	2.3E-05	8.3E-04	6.5E-03	2.4E-02
				5.75	1.3E-08	3.6E-05	1.2E-03	8.6E-03	3.1E-02
				6.00	2.3E-08	5.3E-05	1.6E-03	1.1E-02	3.8E-02

BEST COPY AVAILABLE

PAGE =11

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY				
					1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
LARGE VALVE MAN,CHK,REL	8.90	0.20	0.35	0.50	0.0E+00	4.2E-29	2.1E-24	2.4E-21	3.9E-19
				1.00	4.2E-29	2.4E-21	2.1E-17	7.0E-15	4.6E-13
				1.25	1.8E-26	3.9E-19	2.0E-15	4.6E-13	2.2E-11
				1.50	2.1E-24	2.1E-17	6.6E-14	1.1E-11	4.2E-10
				1.75	9.5E-23	5.0E-16	1.1E-12	1.4E-10	4.4E-09
				2.00	2.4E-21	7.0E-15	1.1E-11	1.1E-09	2.9E-08
				2.25	3.6E-20	6.6E-14	7.9E-11	6.6E-09	1.4E-07
				2.50	3.9E-19	4.6E-13	4.2E-10	2.9E-08	5.6E-07
				2.75	3.2E-18	2.5E-12	1.5E-09	1.1E-07	1.8E-06
				3.00	2.1E-17	1.1E-11	6.6E-09	3.3E-07	5.0E-06
				3.25	1.1E-16	4.2E-11	2.1E-08	9.1E-07	1.2E-05
				3.50	5.0E-16	1.4E-10	5.7E-08	2.2E-06	2.7E-05
				3.75	2.0E-15	4.2E-10	1.4E-07	5.0E-06	5.6E-05
				4.00	7.0E-15	1.1E-09	3.3E-07	1.0E-05	1.1E-04
				4.25	2.3E-14	2.8E-09	7.2E-07	2.0E-05	1.9E-04
				4.50	6.6E-14	6.6E-09	1.4E-06	3.7E-05	3.2E-04
				4.75	1.8E-13	1.4E-08	2.8E-06	6.4E-05	5.2E-04
				5.00	4.6E-13	2.9E-08	5.0E-06	1.1E-04	8.2E-04
				5.25	1.1E-12	5.7E-08	8.7E-06	1.7E-04	1.2E-03
				5.50	2.5E-12	1.1E-07	1.5E-05	2.6E-04	1.8E-03
				5.75	5.4E-12	1.9E-07	2.3E-05	3.9E-04	2.5E-03
				6.00	1.1E-11	3.3E-07	3.7E-05	5.7E-04	3.5E-03
MISC SMALL VALVES	12.50	0.33	0.35	0.50	8.5E-31	5.2E-24	1.9E-20	4.1E-18	2.1E-16
				1.00	5.2E-24	4.1E-18	4.5E-15	4.2E-13	1.1E-11
				1.25	5.2E-22	2.1E-16	1.6E-13	1.1E-11	2.4E-10
				1.50	1.9E-20	4.5E-15	2.4E-12	1.4E-10	2.5E-09
				1.75	3.5E-19	5.3E-14	2.2E-11	1.0E-09	1.6E-08
				2.00	4.1E-18	4.2E-13	1.4E-10	5.5E-09	7.6E-08
				2.25	3.4E-17	2.4E-12	6.5E-10	2.3E-08	2.8E-07
				2.50	2.1E-16	1.1E-11	2.5E-09	7.6E-08	8.5E-07
				2.75	1.1E-15	4.2E-11	8.0E-09	2.2E-07	2.2E-06
				3.00	4.5E-15	1.4E-10	2.3E-08	5.5E-07	5.2E-06
				3.25	1.6E-14	4.0E-10	5.7E-08	1.3E-06	1.1E-05
				3.50	5.3E-14	1.0E-09	1.3E-07	2.7E-06	2.2E-05
				3.75	1.6E-13	2.5E-09	2.8E-07	5.2E-06	4.0E-05
				4.00	4.2E-13	5.5E-09	5.5E-07	9.6E-06	7.0E-05
				4.25	1.0E-12	1.1E-08	1.0E-06	1.7E-05	1.1E-04
				4.50	2.4E-12	2.3E-08	1.9E-06	2.8E-05	1.8E-04
				4.75	5.3E-12	4.2E-08	3.2E-06	4.5E-05	2.8E-04
				5.00	1.1E-11	7.6E-08	5.2E-06	7.0E-05	4.1E-04
				5.25	2.2E-11	1.3E-07	8.3E-06	1.0E-04	5.9E-04
				5.50	4.2E-11	2.2E-07	1.3E-05	1.5E-04	8.2E-04
				5.75	7.7E-11	3.5E-07	1.9E-05	2.2E-04	1.1E-03
				6.00	1.4E-10	5.5E-07	2.8E-05	3.0E-04	1.5E-03

BEST COPY AVAILABLE

PAGE =12

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
LG HORIZONTL VESSELS	3.00	0.30	0.35	0.50	3.3E-19	8.0E-14	4.1E-11	2.1E-09	3.5E-08
				1.00	8.0E-14	2.1E-09	2.9E-07	6.2E-06	5.1E-05
				1.25	2.7E-12	3.5E-08	3.2E-06	5.1E-05	3.3E-04
				1.50	4.1E-11	2.9E-07	1.9E-05	2.4E-04	1.3E-03
				1.75	3.5E-10	1.6E-06	7.8E-05	8.0E-04	3.8E-03
				2.00	2.1E-09	6.2E-06	2.4E-04	2.1E-03	8.6E-03
				2.25	9.6E-09	1.9E-05	6.1E-04	4.5E-03	1.7E-02
				2.50	3.5E-08	5.1E-05	1.3E-03	8.6E-03	2.9E-02
				2.75	1.1E-07	1.2E-04	2.6E-03	1.5E-02	4.5E-02
				3.00	2.9E-07	2.4E-04	4.5E-03	2.3E-02	6.6E-02
				3.25	7.1E-07	4.5E-04	7.4E-03	3.5E-02	9.2E-02
				3.50	1.6E-06	8.0E-04	1.1E-02	4.9E-02	1.2E-01
				3.75	3.2E-06	1.3E-03	1.7E-02	6.6E-02	1.5E-01
				4.00	6.2E-06	2.1E-03	2.3E-02	8.6E-02	1.9E-01
				4.25	1.1E-05	3.1E-03	3.2E-02	1.1E-01	2.3E-01
				4.50	1.9E-05	4.5E-03	4.2E-02	1.3E-01	2.7E-01
				4.75	3.2E-05	6.3E-03	5.3E-02	1.6E-01	3.1E-01
				5.00	5.1E-05	8.6E-03	6.6E-02	1.9E-01	3.5E-01
				5.25	7.8E-05	1.1E-02	8.1E-02	2.2E-01	3.9E-01
				5.50	1.2E-04	1.5E-02	9.7E-02	2.5E-01	4.3E-01
				5.75	1.7E-04	1.9E-02	1.1E-01	2.8E-01	4.6E-01
				6.00	2.4E-04	2.3E-02	1.3E-01	3.1E-01	5.0E-01
SMALL TO MED HX & VESSELS	1.84	0.25	0.35	0.50	2.6E-17	6.4E-12	2.8E-09	1.2E-07	1.7E-06
				1.00	6.4E-12	1.2E-07	1.2E-05	1.9E-04	1.2E-03
				1.25	2.0E-10	1.7E-06	1.1E-04	1.2E-03	6.0E-03
				1.50	2.8E-09	1.2E-05	5.3E-04	4.6E-03	1.8E-02
				1.75	2.3E-08	5.7E-05	1.8E-03	1.2E-02	4.2E-02
				2.00	1.2E-07	1.9E-04	4.6E-03	2.6E-02	7.8E-02
				2.25	5.2E-07	5.3E-04	9.9E-03	4.8E-02	1.3E-01
				2.50	1.7E-06	1.2E-03	1.8E-02	7.8E-02	1.8E-01
				2.75	5.0E-06	2.5E-03	3.1E-02	1.2E-01	2.5E-01
				3.00	1.2E-05	4.6E-03	4.8E-02	1.6E-01	3.2E-01
				3.25	2.8E-05	7.8E-03	7.0E-02	2.1E-01	3.9E-01
				3.50	5.7E-05	1.2E-02	9.6E-02	2.6E-01	4.5E-01
				3.75	1.1E-04	1.8E-02	1.3E-01	3.2E-01	5.2E-01
				4.00	1.9E-04	2.6E-02	1.6E-01	3.7E-01	5.8E-01
				4.25	3.3E-04	3.6E-02	2.0E-01	4.3E-01	6.3E-01
				4.50	5.3E-04	4.8E-02	2.4E-01	4.8E-01	6.8E-01
				4.75	8.2E-04	6.2E-02	2.8E-01	5.3E-01	7.2E-01
				5.00	1.2E-03	7.8E-02	3.2E-01	5.8E-01	7.6E-01
				5.25	1.8E-03	9.6E-02	3.6E-01	6.2E-01	8.0E-01
				5.50	2.5E-03	1.2E-01	4.0E-01	6.6E-01	8.2E-01
				5.75	3.4E-03	1.4E-01	4.4E-01	7.0E-01	8.5E-01
				6.00	4.6E-03	1.6E-01	4.8E-01	7.3E-01	8.7E-01

BEST COPY AVAILABLE

0452

PAGE =13

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY				
					1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
LG VERT STOR VSL-FORMD HD	1.46	0.20	0.35	0.50	2.9E-17	1.5E-11	8.3E-09	4.1E-07	6.0E-06
				1.00	1.5E-11	4.1E-07	4.3E-05	6.6E-04	3.9E-03
				1.25	5.4E-10	6.0E-06	3.7E-04	3.9E-03	1.8E-02
				1.50	8.3E-09	4.3E-05	1.8E-03	1.4E-02	4.9E-02
				1.75	7.1E-08	2.0E-04	5.6E-03	3.4E-02	1.0E-01
				2.00	4.1E-07	6.6E-04	1.4E-02	6.8E-02	1.7E-01
				2.25	1.7E-06	1.8E-03	2.8E-02	1.2E-01	2.6E-01
				2.50	6.0E-06	3.9E-03	4.9E-02	1.7E-01	3.5E-01
				2.75	1.7E-05	7.7E-03	7.8E-02	2.4E-01	4.4E-01
				3.00	4.3E-05	1.4E-02	1.2E-01	3.1E-01	5.3E-01
				3.25	9.7E-05	2.2E-02	1.6E-01	3.9E-01	6.0E-01
				3.50	2.0E-04	3.4E-02	2.1E-01	4.6E-01	6.7E-01
				3.75	3.7E-04	4.9E-02	2.6E-01	5.3E-01	7.3E-01
				4.00	6.6E-04	6.8E-02	3.1E-01	5.9E-01	7.8E-01
				4.25	1.1E-03	9.0E-02	3.7E-01	6.5E-01	8.2E-01
				4.50	1.8E-03	1.2E-01	4.2E-01	7.0E-01	8.6E-01
				4.75	2.7E-03	1.4E-01	4.8E-01	7.4E-01	8.9E-01
				5.00	3.9E-03	1.7E-01	5.3E-01	7.8E-01	9.1E-01
				5.25	5.6E-03	2.1E-01	5.7E-01	8.2E-01	9.3E-01
				5.50	7.7E-03	2.4E-01	6.2E-01	8.5E-01	9.4E-01
				5.75	1.0E-02	2.8E-01	6.6E-01	8.7E-01	9.5E-01
				6.00	1.4E-02	3.1E-01	7.0E-01	8.9E-01	9.6E-01
LG VERT FLAT BOTTOM TANK	0.45	0.25	0.45	0.50	9.8E-06	1.7E-03	1.6E-02	5.8E-02	1.3E-01
				1.00	1.7E-03	5.8E-02	2.2E-01	4.1E-01	5.8E-01
				1.25	6.4E-03	1.3E-01	3.6E-01	5.8E-01	7.4E-01
				1.50	1.6E-02	2.2E-01	5.0E-01	7.1E-01	8.4E-01
				1.75	3.3E-02	3.1E-01	6.2E-01	8.0E-01	9.0E-01
				2.00	5.8E-02	4.1E-01	7.1E-01	8.7E-01	9.4E-01
				2.25	8.9E-02	5.0E-01	7.8E-01	9.1E-01	9.6E-01
				2.50	1.3E-01	5.8E-01	8.4E-01	9.4E-01	9.8E-01
				2.75	1.7E-01	6.5E-01	8.8E-01	9.6E-01	9.8E-01
				3.00	2.2E-01	7.1E-01	9.1E-01	9.7E-01	9.9E-01
				3.25	2.6E-01	7.6E-01	9.3E-01	9.8E-01	9.9E-01
				3.50	3.1E-01	8.0E-01	9.5E-01	9.9E-01	1.0E+00
				3.75	3.6E-01	8.4E-01	9.6E-01	9.9E-01	1.0E+00
				4.00	4.1E-01	8.7E-01	9.7E-01	9.9E-01	1.0E+00
				4.25	4.6E-01	8.9E-01	9.8E-01	1.0E+00	1.0E+00
				4.50	5.0E-01	9.1E-01	9.8E-01	1.0E+00	1.0E+00
				4.75	5.4E-01	9.3E-01	9.9E-01	1.0E+00	1.0E+00
				5.00	5.8E-01	9.4E-01	9.9E-01	1.0E+00	1.0E+00
				5.25	6.2E-01	9.5E-01	9.9E-01	1.0E+00	1.0E+00
				5.50	6.5E-01	9.6E-01	9.9E-01	1.0E+00	1.0E+00
				5.75	6.8E-01	9.7E-01	1.0E+00	1.0E+00	1.0E+00
				6.00	7.1E-01	9.7E-01	1.0E+00	1.0E+00	1.0E+00

BEST COPY AVAILABLE

PAGE =14

SSE =0.10

COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
COND STORAGE TANK	0.81	0.28	0.45	0.50	7.4E-08	4.0E-05	7.3E-04	4.2E-03	1.3E-02
				1.00	4.0E-05	4.2E-03	3.0E-02	9.2E-02	1.8E-01
				1.25	2.1E-04	1.3E-02	7.3E-02	1.8E-01	3.1E-01
				1.50	7.3E-04	3.0E-02	1.3E-01	2.9E-01	4.4E-01
				1.75	1.9E-03	5.7E-02	2.1E-01	3.9E-01	5.6E-01
				2.00	4.2E-03	9.2E-02	2.9E-01	4.9E-01	6.5E-01
				2.25	7.8E-03	1.3E-01	3.7E-01	5.8E-01	7.3E-01
				2.50	1.3E-02	1.8E-01	4.4E-01	6.5E-01	7.9E-01
				2.75	2.1E-02	2.3E-01	5.1E-01	7.2E-01	8.4E-01
				3.00	3.0E-02	2.9E-01	5.8E-01	7.7E-01	8.8E-01
				3.25	4.2E-02	3.4E-01	6.4E-01	8.1E-01	9.1E-01
				3.50	5.7E-02	3.9E-01	6.9E-01	8.5E-01	9.3E-01
				3.75	7.3E-02	4.4E-01	7.3E-01	8.8E-01	9.4E-01
				4.00	9.2E-02	4.9E-01	7.7E-01	9.0E-01	9.6E-01
				4.25	1.1E-01	5.4E-01	8.0E-01	9.2E-01	9.7E-01
				4.50	1.3E-01	5.8E-01	8.3E-01	9.3E-01	9.7E-01
				4.75	1.6E-01	6.2E-01	8.6E-01	9.5E-01	9.8E-01
				5.00	1.8E-01	6.5E-01	8.8E-01	9.6E-01	9.8E-01
				5.25	2.1E-01	6.9E-01	9.0E-01	9.6E-01	9.9E-01
				5.50	2.3E-01	7.2E-01	9.1E-01	9.7E-01	9.9E-01
				5.75	2.6E-01	7.5E-01	9.2E-01	9.8E-01	9.9E-01
				6.00	2.9E-01	7.7E-01	9.3E-01	9.8E-01	9.9E-01
AIR HANDLING UNITS	6.90	0.27	0.45	0.50	3.0E-21	3.6E-16	1.5E-13	7.5E-12	1.3E-10
				1.00	3.6E-16	7.5E-12	1.2E-09	2.9E-08	2.8E-07
				1.25	1.1E-14	1.3E-10	1.4E-08	2.8E-07	2.4E-06
				1.50	1.5E-13	1.2E-09	9.8E-08	1.6E-06	1.2E-05
				1.75	1.3E-12	6.7E-09	4.6E-07	6.5E-06	4.2E-05
				2.00	7.5E-12	2.9E-08	1.6E-06	2.0E-05	1.2E-04
				2.25	3.4E-11	9.8E-08	4.7E-06	5.2E-05	2.7E-04
				2.50	1.3E-10	2.8E-07	1.2E-05	1.2E-04	5.7E-04
				2.75	4.1E-10	7.2E-07	2.6E-05	2.3E-04	1.1E-03
				3.00	1.2E-09	1.6E-06	5.2E-05	4.3E-04	1.8E-03
				3.25	2.9E-09	3.4E-06	9.6E-05	7.3E-04	2.9E-03
				3.50	6.7E-09	6.5E-06	1.7E-04	1.2E-03	4.5E-03
				3.75	1.4E-08	1.2E-05	2.7E-04	1.8E-03	6.5E-03
				4.00	2.9E-08	2.0E-05	4.3E-04	2.7E-03	9.1E-03
				4.25	5.4E-08	3.3E-05	6.5E-04	3.8E-03	1.2E-02
				4.50	9.8E-08	5.2E-05	9.4E-04	5.2E-03	1.6E-02
				4.75	1.7E-07	7.9E-05	1.3E-03	7.0E-03	2.1E-02
				5.00	2.8E-07	1.2E-04	1.8E-03	9.1E-03	2.7E-02
				5.25	4.6E-07	1.7E-04	2.4E-03	1.2E-02	3.3E-02
				5.50	7.2E-07	2.3E-04	3.2E-03	1.5E-02	4.0E-02
				5.75	1.1E-06	3.2E-04	4.1E-03	1.8E-02	4.8E-02
				6.00	1.6E-06	4.3E-04	5.2E-03	2.2E-02	5.6E-02

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COMPONENT	MEDIAN FRAGIL	LOG SD FRAGIL	LOG SD RESP	MULT FACTOR	SEISMIC FAILURE PROBABILITY 1 SSE	2 SSE	3 SSE	4 SSE	5 SSE
GAS TURBINE GENERATOR	12.10	0.27	0.25	0.50	0.0E+00	0.0E+00	4.1E-33	3.6E-29	2.7E-26
				1.00	0.0E+00	3.6E-29	4.7E-24	9.7E-21	2.4E-18
				1.25	9.4E-36	2.7E-26	1.8E-21	2.4E-18	4.0E-16
				1.50	4.1E-33	4.7E-24	1.8E-19	1.6E-16	2.1E-14
				1.75	5.7E-31	3.0E-22	7.5E-18	4.8E-15	4.7E-13
				2.00	3.6E-29	9.7E-21	1.6E-16	7.8E-14	6.2E-12
				2.25	1.2E-27	1.8E-19	2.2E-15	8.2E-13	5.4E-11
				2.50	2.7E-26	2.4E-18	2.1E-14	6.2E-12	3.4E-10
				2.75	4.2E-25	2.2E-17	1.5E-13	3.6E-11	1.7E-09
				3.00	4.7E-24	1.6E-16	8.2E-13	1.7E-10	7.0E-09
				3.25	4.2E-23	9.6E-16	3.8E-12	6.7E-10	2.4E-08
				3.50	3.0E-22	4.8E-15	1.5E-11	2.3E-09	7.4E-08
				3.75	1.8E-21	2.1E-14	5.4E-11	7.0E-09	2.0E-07
				4.00	9.7E-21	7.8E-14	1.7E-10	1.9E-08	5.0E-07
				4.25	4.5E-20	2.7E-13	4.8E-10	4.8E-08	1.1E-06
				4.50	1.8E-19	8.2E-13	1.3E-09	1.1E-07	2.4E-06
				4.75	6.9E-19	2.3E-12	3.1E-09	2.4E-07	4.8E-06
				5.00	2.4E-18	6.2E-12	7.0E-09	5.0E-07	9.1E-06
				5.25	7.5E-18	1.5E-11	1.5E-08	9.7E-07	1.6E-05
				5.50	2.2E-17	3.6E-11	3.1E-08	1.8E-06	2.8E-05
				5.75	6.2E-17	8.0E-11	6.0E-08	3.2E-06	4.7E-05
				6.00	1.6E-16	1.7E-10	1.1E-07	5.5E-06	7.5E-05

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Appendix C.2

Review of Site-Specific Fragilities Derived from 19 Recent PRAs

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6160 Birch Street, Newport Beach, Calif 92660 (714) 833-7552

NTS/SMA 12607.09

January 21, 1985

Dr. M. P. Bohn
SANDIA NATIONAL LABORATORY
P.O. Box 5800
Albuquerque, New Mexico 87115

Dear Dr. Bohn:

We have completed a comprehensive review of the seismic fragility levels for mechanical and electrical equipment generally found in nuclear power plant facilities. The evaluation centered upon the 31 specific categories of equipment shown in Table 1. This letter report presents a description of the data base used to establish the median fragility levels and their variabilities, a discussion of the process used to obtain the final statistical parameters, and a discussion of the significant differences between the results of this study and fragility levels previously establish during the SSMRP.

SMA has provided the fragility description for mechanical and electrical equipment in support of some 20 Probabilistic Risk Assessments (PRAs) of existing nuclear power plant facilities during the past four years. The calculated fragility levels for these PRAs have been based upon either plant specific stress levels obtained from code analysis reports, SQRT forms, and/or qualification test reports or upon a generic development using either knowledge of general equipment design practices or data from generic equipment dynamic tests. For the current study, all those components whose fragility was based upon well defined plant specific design stress information or dynamic qualification test data were included in the data base. All such components were placed in one of the defined categories and their floor level seismic acceleration capacity was computed based upon a recognition of the conservatism or unconservatism present in the seismic analysis of the component and the actual failure point of the component. The data base which can be obtained from plant specific seismic analyses is still somewhat limited such that the number of samples in each category ranges from 2 to 51. However, since the emphasis in the PRA evaluations involves a search for "weak links" or low capacity components within a safety related system, it is believed that the data base is at least slightly conservatively biased.

Table 1 presents the statistical parameters associated with the floor level seismic acceleration, capacities determined for the components in each category. The calculated median capacity values for each component

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were used in the statistical evaluation. The results presented in Table 1 are defined as follows:

Sa_{Min} = Minimum median acceleration capacity value for all samples within the category

Sa_{Max} = Maximum median acceleration capacity value for all samples within the category

$\overline{Sa}$ = Mean value of acceleration capacity

σ = Standard deviation

δ = Coefficient of variation ($\delta = \sigma/\overline{Sa}$)

β_U = Uncertainty variability of the sample distribution

$$\beta_U = \sqrt{\ln\,(1 + \delta^2)}$$

β_R = Randomness variability of the sample distribution

$$\beta_R = \frac{\sum_{i=1}^{N} (\beta_{R_i})}{N}$$

$\check{Sa}$ = Median value of acceleration capacity

$$\check{Sa} = \overline{Sa}\, e^{-1/2\, \beta_U^2}$$

HCLF = High Confidence - Low Frequency capacity value representing 95% confidence of less than 5% frequency of failure

$$\text{HCLF} = \check{Sa}\, e^{-1.65\,(\beta_R + \beta_U)}$$

The statistical median values and their variabilities were originally computed from the equations given above for all samples within each category. However, in many cases the uncertainty variability (β_U) was found to be inordinately large. By plotting the distribution of seismic acceleration capacity for each category, as shown in Figure 1 for vertical and horizontal pumps, it was found that high capacity outliers seriously distorted the distributions due to the limited number of samples in each category. Since it is of interest in this study to define the lower half of the fragility curve as accurately as possible (the upper half is not of concern), and

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since a possibility exists that the data for these high capacity samples did not define the lowest capacity failure mode for the item, the extremely high capacity outliers were removed from the sample set and the statistical median values and their variabilities were recomputed. The values shown in Table 1 reflect results of the latter analysis. The high confidence-low frequency value calculated for each category was then compared with the experience data of similar components subjected to real earthquakes gathered by the Seismic Qualification Utility Group (SQUG). It was found in all cases that the calculated HCLF values exceeded earthquake accelerations at which no failures were found to occur. It was also judged that the resulting median capacity values ($\check{S}_a$) were realistic in magnitude and bore reasonable comparisons between the various categories. It should be noted that the uncertainty variability (β_U) given in Table 1 is based upon the variation of the median seismic capacity values of the samples in each category and is not related to the uncertainty in the calculation of the median values for each component. As a result, where the sample size is small (i.e., Safety Relief Valves or Compressors) or where the range from the minimum to maximum value is small (i.e., Motor Control Center -Trip Failure) the tabulated β_U values should probably in increased.

A comparison was made between the results of this study and the median seismic capacities recommended for the various categories from the SSMRP effort. In most cases there was reasonable agreement. However, there were a few notable exceptions which will be discussed. The median values and uncertainty variabilities from this study and the SSMRP recommended values are summarized in Table 2.

SSMRP recommended a seismic capacity for diesel generators of 0.65g. During the course of the PRA studies the diesel generators were reviewed in detail for most plants. The capacities were found to generally be well in excess of 1.0g. The results of this study indicate that a median capacity on the order of 1.95g is appropriate for diesel generators.

SSMRP recommended acceleration capacities of 1.84g and 2.01g for small tanks or accumulators and flat bottom storage tanks, respectively. In general small accumulators exhibit very high capacities in excess of 10g. This study indicated a capacity of 17.08g, which may be somewhat high. However, it should be noted that a very high capacity outlier was eliminated from the data set even to achieve the tabulated value. In contrast flat bottom storage tanks which are typically located outside have seldom been found to exhibit an acceleration capacity in excess of 1.0g. It is believed that the SSMRP value should be decreased significantly. The sample set from this study indicates that a median capacity of 0.45g is appropriate for flat bottom storage tanks.

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SSMRP recommended acceleration capacities of 9.84g and 4.83g for small motor operated valves (≤ 4 inch) and large motor operated valves (>4 inch), respectively. It is believed that these should essentially be reversed. Numerous studies have shown that the large motor operators are more critical for small valves whose section modulus is small. This study also shows this trend and indicates that seismic capacities of 4.68g and 11.04g are appropriate for small and large motor operated valves, respectively.

Although the sample is small, the seismic capacity of inverters is indicated to be on the order of 2.01g in contrast to the 15.6g value recommended by SSMRP. It is believed that the SSMRP value was based upon a beefed up unit specificly designed for use in a high seismic plant.

Finally, more recent data indicates that the median seismic capacity of ceramic insulators should be increased to 0.30g over the 0.20g originally specified by the SSMRP.

Some care should be taken in the interpretation and use of the results of this study. It should be noted that median capacity values for a wide range of component designs are potentially included in this data set. As a result, the plant risk for a given nuclear facility may be governed by a component whose capacity is less than the median value for the appropriate category. Therefore, for initial screening studies employed to determine those components whose failure would be the most significant contributors to plant risk, it is recommended that a seismic capacity value somewhat less than the median value be specified. For example one might assume a median value equal to the minus one log standard deviation from the data base.

$$Sa = \check{Sa}\, e^{-\beta_U}$$

However, if such a reduced value of seismic capacity is specified, the uncertainty variability (β_U) should also be reduced accordingly such that the HCLF of the data base would be unchanged. The reduced uncertainty variability would then be calculated as

$$\beta_U' = \frac{1}{-1.65}\left[\ln\left(\frac{HCLF}{Sa}\right)\right] - \beta_R$$

where Sa is the specified capacity value. Such an approach could identify those components which should be evaluated in more detail to assure acceptably low plant risk.

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If you have any questions concerning the results of this study, please do not hesitate to give me a call at the number listed above. We thank you for the opportunity to be of service to you.

Sincerely,

NTS/STRUCTURAL MECHANICS ASSOCIATES

Thomas R. Kipp

Thomas R. Kipp
Project Manager

cc: J. J. Johnson

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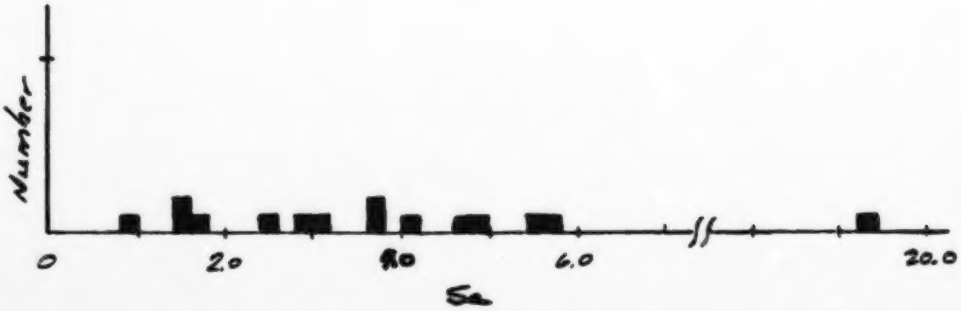
0461

VERTICAL PUMPS (15 Samples)



HORIZONTAL PUMPS (51 Samples)

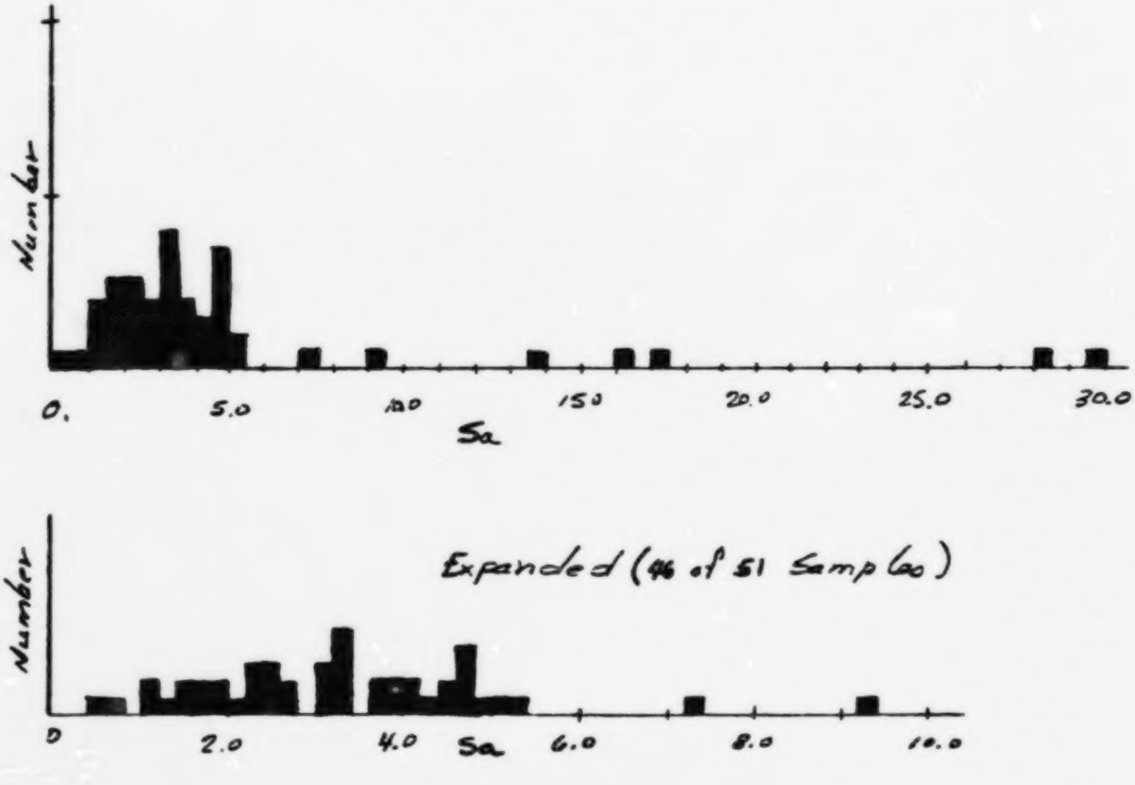


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Figure 1 Seismic Capacity Distributions For Vertical and Horizontal Pumps

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TABLE 1 CAPACITIES AND VARIABILITIES FOR PLANT SPECIFIC EQUIPMENT CATEGORIES

CATEGORY	NO. OF SAMPLES	FREQUENCY RANGE	$S_{a\,MIN}$	$S_{a\,MAX}$	$\bar{S}_a$	σ	δ	$\check{S}_a$	β_U	β_R
Main Steam ISO Valve	5	5 - 20 Hz	3.02	10.80	7.15	3.00	0.42	6.59	0.40	0.24
Diesel Generator	10	11 - Rigid	0.75	3.74	2.13	0.95	0.45	1.95	0.43	0.16
Pumps (Long Column)	7	7 - Rigid	0.97	5.82	2.99	1.81	0.61	2.55	0.56	0.17
Pumps (Vertical)	14	7 - Rigid	0.98	5.67	3.29	1.56	0.48	2.97	0.45	0.21
Pumps (Horizontal)	44	7 - Rigid	0.49	5.37	3.03	1.29	0.43	2.79	0.41	0.15
Heat Exchangers (Vertical)	9	6 - Rigid	1.28	4.80	2.43	1.02	0.42	2.24	0.40	0.21
Heat Exchangers (Horizontal)	11	14 - Rigid	1.11	9.40	4.55	3.20	0.70	3.72	0.63	0.16
Tanks (Small Accumulators)	4	22 - Rigid	12.5	28.6	18.31	7.09	0.39	17.08	0.37	0.19
Tanks (Flat Bottom)	11	5 - 15	0.16	0.88	0.52	0.29	0.56	0.45	0.52	0.26
Tanks (Vertical)	13	3 - Rigid	1.44	7.50	3.66	1.98	0.54	3.21	0.51	0.15
Tanks (Horizontal)	14	8 - Rigid	1.22	6.56	3.35	1.60	0.48	3.02	0.45	0.19
Fan Coolers	8	4 - 22	1.86	16.80	8.30	5.56	0.67	6.90	0.61	0.25
Valves (Safety Relief)	4	5 - 20(Piping)	7.30	9.17	8.48	0.86	0.10	8.44	0.10	0.27
Valves (A.O & M.O ≤ 4")	9	5 - 20(Piping)	1.73	9.97	5.33	2.90	0.54	4.68	0.51	0.19
Valves (A.O & M.O > 4")	47	5 - 20(Piping)	2.31	27.0	12.49	6.59	0.53	11.04	0.50	0.21
Compressors	2	10 - 20	2.36	3.09	2.73	0.52	0.19	2.68	0.19	0.12
Turbines (HPCI & RCIC)	6	15 - Rigid	2.92	6.09	5.21	1.22	0.23	5.10	0.23	0.19
Electrical Panels & Boards (Chatter)	41	2 - 20	1.60	10.23	4.71	2.50	0.53	4.19	0.50	0.19
Electrical Panels & Boards (Trip)	10	2 - 20	5.54	13.53	10.53	4.72	0.45	9.60	0.43	0.25
Switchgear (Chatter)	11	4 - 10	1.94	7.35	5.13	1.94	0.38	4.79	0.37	0.17
Switchgear (Trip)	6	4 - 10	4.58	9.98	6.70	1.98	0.29	6.43	0.29	0.21
Switchgear (Structural)	11	4 - 10	1.50	14.7	6.29	5.13	0.82	4.87	0.71	0.24
Motor Control Centers (Chatter)	11	4 - 18	1.79	8.53	4.83	2.34	0.49	4.35	0.46	0.16
Motor Control Centers (Trip)	5	4 - 10	6.85	8.00	7.29	0.45	0.06	7.27	0.06	0.21
Batteries and Racks	10	3 - Rigid	1.21	5.60	2.98	1.43	0.48	2.68	0.46	0.23
Battery Chargers	6	5 - 20	1.76	9.35	3.68	1.70	0.46	3.34	0.44	0.24
Transformers	6	5 - 20	5.49	14.25	9.32	3.34	0.36	8.77	0.35	0.26
Switches	4	13 - 20	1.73	6.46	3.48	2.26	0.65	2.92	0.59	0.25
Inverters	2	8 - 14	1.67	2.52	2.10	0.60	0.29	2.01	0.28	0.14
Instrumentation	5	Rigid	6.84	14.97	11.23	3.83	0.34	10.63	0.33	0.34
Ceramic Insulators	-	-	-	-	-	-	-	0.30	0.37	0.30

0463

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TABLE 2 COMPARISON OF RESULTS WITH SSMRP RECOMMENDATIONS

CATEGORY	Current Study		SSMRP	
	$\check{S}a$	β_U	$\check{S}a$	β_U
Main Steam 150 Valve	6.59	0.40	-	-
Diesel Generator	1.95	0.43	0.65	0.40
Pumps (Long Column)	2.55	0.56	-	-
Pumps (Vertical)	2.97	0.45	2.21	0.39
Pumps (Horizontal)	2.79	0.41	3.19	0.34
Heat Exchangers (Vertical)	2.24	0.40	-	-
Heat Exchangers (Horizontal)	3.72	0.63	-	-
Tanks (Small Accumulators)	17.08	0.37	1.84	0.51
Tanks (Flat Bottom)	0.45	0.52	2.01	0.38
Tanks (Vertical)	3.21	0.51	1.46	0.56
Tanks (Horizontal)	3.02	0.45	3.91	0.61
Fan Coolers	6.90	0.61	2.24	0.41
Valves (Safety Relief)	8.44	0.10	8.90	0.40
Valves (A.O & M.O≤ 4")	4.68	0.51	9.84	0.65
Valves (A.O & M.O >4")	11.04	0.50	4.83	0.65
Compressors	2.68	0.19	3.19	0.34
Turbines (HPCI & RCIC)	5.10	0.23	-	-
Electrical Panels & Boards (Chatter)	4.19	0.50	6.00	0.89
Electrical Panels & Boards (Trip)	9.60	0.43	7.63	0.88
Switchgear (Chatter)	4.79	0.37	2.33	0.81
Switchgear (Trip)	6.43	0.29	-	-
Switchgear (Structural)	4.87	0.71	11.50	0.88
Motor Control Centers (Chatter)	4.35	0.46	-	-
Motor Control Centers (Trip)	7.27	0.06	7.63	0.88
Batteries and Racks	2.68	0.46	2.29	0.50
Battery Chargers	3.34	0.44	-	-
Transformers	8.77	0.35	2.78	0.41
Switches	2.92	0.59	-	-
Inverters	2.01	0.28	15.60	0.44
Instrumentation	10.63	0.33	7.66	0.40
Ceramic Insulators	0.30	0.37	0.20	-

0464

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APPENDIX D

FIRE RISK ANALYSIS FOR THE ST. LUCIE NUCLEAR POWER PLANT

by

Wallace T. Wheelis
Sandia National Laboratories

TABLE OF CONTENTS

1. ANALYSIS METHODS AND ASSUMPTIONS 3 467

1.1 Introduction 3 467

1.2 Scope 3 467

1.3 Analysis Methodology 4 468

1.3.1 Initial Screening for Potentially Important Fire Areas 5 469
1.3.2 Plant Visit 5 469
1.3.3 Quantification 6 470
1.3.4 Proposed Modifications 8 472

1.4 Assumptions 8 472

2. APPLICATION OF METHODOLOGY 14 478

2.1 Initial Screening to Determine Potentially Important Fire Areas 14 478
2.2 Quantification of Selected Fire Zones 21 485

2.2.1 Fire Zone 57: Battery Charger, Inverter, and Cable Spreading Room 21 485
2.2.2 Fire Zone 70: Control Room 30 494

3. CONCLUSIONS AND INSIGHTS 31 495

4. REFERENCES 34 498

5. TABLES 35 499

1. ANALYSIS METHODS AND ASSUMPTIONS

1.1 Introduction

Based on plant operating experience over the last 20 years, it has been observed that nuclear power plants will have three to four significant fires over their operating lifetimes (i.e., on the order of one fire every ten years or a frequency of .1/Rx-yr). Previous probabilistic risk assessments (PRAs) have shown that fires are a significant contributor to the overall core melt probability, contributing anywhere from 7% to 50% of the total (considering contributions from internal, seismic, flood, fire and other events). Because of the relatively high core melt contribution, fires need to be examined in more detail.

Large strides have been made in fire protection for nuclear power plants in recent years. In particular, 10 CFR 50, Appendix R and Branch Technical Positions 9.5-1 have led to improved fire protection measures at nuclear power plants. However, even with these improvements, there is still a residual risk from fires. These residual risks have not yet been quantified with the exception of those plants that have performed fire related PRAs. The purpose of this section is to quantify the risks due to fires at the St. Lucie Unit 1 Nuclear Power Plant.

1.2 Scope

a. This analysis will evaluate for the St. Lucie Unit 1 Nuclear Power Plant the residual risk from fires that could affect decay heat removal systems. The results of this analysis, the quantified risk and any modifications that may be proposed, will be compared to risks from other initiating events with the modifications evaluated for risk reduction and cost. This "value-impact" analysis will then ultimately determine the optimum set of modifications to reach the desired reduction in risk.

b. The analysis is not intended to be a comprehensive fire PRA. An analysis of this sort would be beyond the scope of TAP A-45. The intent of this analysis is to quantify significant fire threats to a plant. In doing so, existing procedures used in published fire PRAs will be employed, but simplifying assumptions will be made to allow the analysis to proceed without requiring the level of detail required for these other fire PRAs. In that respect, this analysis embodies the basic approaches used by previous fire PRAs but is by necessity a scaled down version of these PRAs.

c. Wherever possible, data from previous fire PRAs will be used. Since this analysis embodies what has been done in previous PRAs, use of this data will provide consistency in how the data is used and allow some comparison between the analyses. The data is based on that found in the Limerick [1], Millstone 3 [2] and Seabrook [3] fire PRAs (three of the most recent fire PRAs).

d. Because this analysis is a scaled down version of a fire PRA, quantification of the risk due to a fire will be presented as a point estimate. Uncertainty values will not be formally determined. However, insights into the potential major areas of uncertainty will be identified.

e. The systems to be examined and quantified are based on the event and fault trees developed by the systems analysts, and thus are constrained by the level of detail of the systems analysis effort. In a similar fashion, readily available plant documentation determines the level of physical detail used in this analysis. Documentation obtained for this effort is treated as accurate and up to date unless otherwise identified. No attempt is made to physically verify the locations of selected cables referenced in this documentation.

f. A screening cutoff of 1.0E-06 for fire-induced core melt probability was used to determine if further quantification efforts in an area were warranted. This was based on internal events analyses that showed core melt probabilities from internal events alone were on the order of 1.0E-04 so that fires less than 1.0E-06 would be insignificant.

1.3 Analysis Methodology

The approach used in determining the risk from a fire at the St. Lucie Unit 1 Nuclear Power Plant used four basic steps:

a. Initial screening for potentially important fire areas
b. Plant visit for verification
c. Quantification
d. Proposed modifications to reduce the fire risk in affected areas

1.3.1 Initial Screening for Potentially Important Fire Areas

The purpose of this step was to determine which areas in the plant had "enough" front line or support systems in their boundaries such that fire damage could invalidate all the success branches on the transient event tree. At this stage of the screening, the capability to remotely shut down the plant via local or remote stations was not considered.

The sequence of steps taken during this first screening were as follows:

0468

a. Internal event analysis data was obtained as the starting point. The specific data included event trees, fault trees, and simplified piping diagrams used in constructing the fault trees.

b. A transient event tree was examined to determine those front line systems or combinations of systems that would lead to an 'OK' state on the event tree. Also obtained from the event tree were the success criteria of front line systems.

c. The next step was to identify the support systems required for the front line systems identified above. Support systems included cooling systems and AC/DC power requirements. The success criteria for the support systems were obtained from the systems analysis. For all systems except the power conversion system, the fault trees for the front line systems were used to determine the support systems. For the power conversion system, plant schematics and piping diagrams were used to determine required support systems.

d. Once the systems of interest were identified, all available documentation (principally 10 CFR 50, Appendix R submittals, FSARs, and plant supplied information and drawings) was analyzed to determine those areas where "enough" components could be disabled by a fire. Based on this screening, a small set of areas were left for further analysis.

To this point in the analysis, everything was based on a review of available documentation.

1.3.2 Plant Visit

A visit was made to the plant with the following objectives:

a. Based on the initial screening, the general location of cables and components of the systems of interest were known. The next step was to visit the plant and see the physical arrangements in each of these areas. The plant visit also provided a means to determine locations of major fuel sources and some estimate of the x,y,z coordinates of components and cables. This location information was necessary for the quantification step.

b. The second purpose of the plant visit was to confirm with plant personnel that the documentation that was being used was in fact the best available information and to obtain clarification about any questions that might have arisen in a review of the documentation.

1.3.3 Quantification

For those areas remaining after the initial screening, the next step was to run a deterministic fire code (COMPBRN) [4] to ascertain if an exposure fire could damage the components/cables of interest in the area. The types of exposure fires utilized were a trash can fire and an acetone spill of 10 gallons. Based on the code results, if damage occurred then the analysis continued. If no damage occurred, the analysis was complete. The code calculations not only included cable failure from ignition, but also failure prior to ignition resulting from melting of the internal conductors. The melting is a function of temperature and time at temperature. Cable damage temperatures are discussed in the assumptions section.

Other factors were also considered at this stage to determine if further quantification of fire risk in a particular area should continue. Because control rooms are continually manned, the chance that a fire would not be suppressed prior to causing damage to enough of the systems of interest is less than for other, non-occupied areas with the same basic fuel load. However, if "enough" of the systems of interest were bunched together, i.e., on a single control board section, a fire could potentially disable these safety systems. If this arrangement occurred, the control room fire risk was quantified; otherwise, it was not. Additionally, cables in any area that had a thermal blanket (like KAOWOOL) were assumed to survive a fire equal to the blanket manufacturer's rating.

An event tree was utilized to estimate the risk of core melt from a fire in those areas identified in the initial screening and not eliminated by code calculations. The basic elements of the event tree were as follows:

a. Fire occurrence frequency for the area of interest.

b. Suppression system probability (both for automatic and manual suppression activity).

c. Operator error and random failures of unaffected systems from a remote shutdown location.

Each of these elements are discussed below:

a. Fire occurrence frequency for the area of interest

The fire frequency for each area was based on historical fire occurrence data found in previous fire PRAs. Table D.1 summarizes this data. Several specific rooms in nuclear power plants have their own fire frequency values, while other areas are only included in a building frequency. Data for specific rooms was used "as is " in this analysis. For rooms not included in the listing, a weighting factor was used to reduce a building frequency to a room frequency. Depending on the available information, either a cable weighting or an area weighting approach was used. For St. Lucie Unit 1 an area weighting approach was used. The area weighting method is a ratio of the individual room floor area to that of the building it is in. (The cable weighting method is a ratio of the the weight of cables in the room of interest to the weight of cables found in the rest of the building). Both of these methods have been used in previous fire PRAs.

The presence and location of an automatic fire detection system (such as a smoke detector) was not explicitly considered and factored into the analysis since in almost all cases plants have installed detectors throughout the plant. A probability was not assigned for detector failure because the detectors are highly reliable (on the order of 95 to 99%) and the probability of manual suppression incorporates detector response time (i.e. it was assumed that given a fire it would be detected). This step is consistent with prior fire PRAs.

b. Suppression system probability

Both automatic (i.e., halon, CO_2, water) and manual suppression systems were considered. Automatic suppression systems were assumed to actuate during the fire and immediately extinguish the fire. The reliabilities for the various automatic systems are given in Table D.2 and represent the probability that an automatic system malfunctions (i.e. does not actuate). Manual suppression probability was based on the time that the computer code predicted for the fire to damage the components of interest. Table D.3 lists the non-suppression probabilities. (Note that these probabilities do not include suppression damage, but rather just the probability that the fire can be put out in a given time frame.)

c. Operator error and random failures of unaffected systems from a remote shutdown location.

Any unaffected systems that could be either locally or remotely used to provide decay heat removal were evaluated. For fires that required the use of a remote shutdown location, the chance of incorrect operation was evaluated. This human error probability was taken to be .1, based on Seabrook [3] PRA data for the conditional frequency that the operators will not do the right thing at auxiliary shutdown panels. Next, the reliability of the remaining decay heat removal system(s) was evaluated using the fault trees for the plant. Values for the random failure of a single train were on the order of 1.0E-02/demand.

As an additional step, which should be treated only as a bounding calculation, failure of systems due to manual

0471

suppression damage was evaluated. Any components not affected by the fire were assumed to fail when manual suppression occurred. For this situation, the core melt probabilities were requantified. Damage from automatic suppression systems in the room or inadvertent operation in other locations was not assessed. (It is recognized that this is a valid concern, and that automatic suppression damage potentially could increase the overall core melt probability, however the data bases and experimental data required to incorporate this concern do not presently exist and the generation of this type of data is clearly beyond the scope of the TAP A-45 analysis.)

1.3.4 Proposed Modifications

Based on the quantification steps, engineering modifications were proposed for those areas which have a fire- induced core melt probability contribution of greater than 1.0E-06/Rx yr. The proposed modifications were of the following nature:

a. Installation of automatic suppression.

b. Cable wrapping (only applicable if the cables being wrapped would not overheat from internal heat generation).

c. Construction of barriers to isolate portions of an area.

d. Automation of manual systems to improve their reliability.

e. Rerouting of affected cables or physically moving redundant components to other locations.

f. Procedural changes

For any proposed modification, a core melt probability was recalculated to indicate how much the core melt probability could be reduced. The quantification was performed by modifying the event tree to reflect the modification.

The proposed modifications were supplied to an architectural engineering firm for cost evaluation during a second plant visit. A "value-impact" analysis considering the core melt contributions from fires and other external and internal events was used to determine the "best modification" in an overall sense.

1.4 Assumptions

Most of the assumptions are based on insights gained from previous fire PRAs or from the historical fire data base. The assumptions are as follows:

a. Only fires in a single area (room) are considered, i.e. no room to room issues are addressed.

0472

From a review of previous PRAs, only single areas (rooms) have been significant contributors to the core melt probability from a fire. Although this procedure neglects certain equipment vulnerabilities (smoke and suppression system actuation outside the fire area) and the effects of ventilation system operation, these factors are assumed to be of less risk significance than single area fires.

Because no room to room issues are being considered, barriers between adjacent areas are assumed to remain intact during a fire, limiting the fire to only one area. Thus, the effectiveness of an area's fire barrier, either from a deterministic or probabilistic standpoint, will not be evaluated. It is assumed that the fire barriers, including walls, doors, dampers, and penetrations, have been properly constructed to ensure their integrity in a fire.

b. Fire frequency and suppression reliability is based on available industry data.

Existing data was used to the extent possible in determining the threat of fire to a plant. Fire frequency data has been collected for all the fire PRAs performed. This data has been typically used as generic data that is modified to reflect the actual fire experiences at the plant. For this analysis, the fire frequency used was the mean value of the generic data; the plant's fire history was used to modify this generic data. Suppression system data was based on nuclear insurers data for installed automatic systems [2] and on data from work done for the HTGR fire methodology PRA.[5] This data has been used by almost all previous fire PRAs when evaluating manual suppression probabilities.

c. Two transient combustible exposure fires were assumed to bound all transient and electrically-initiated fires.

The following exposure fires were used in the computer calculations:

1. A trash can size container of refuse with an energy output of approximately 500 kW for 15 minutes. The total heat released was approximately 4.5E+08 Joules. For transient combustibles, such as trash, survey information from 35 Inspection and Enforcement (I&E) inspectors indicated that 30 gallon trash containers are the most likely size of containers found at nuclear power plants [6]. Test data for fires of this size indicates that an energy output of 500 kW for 15 minutes represents what could reasonably be expected.

2. A pool fire (which most of the fire codes are designed to handle in a straightforward way and are benchmarked against) was used as a second transient source. A spill of 0473

approximately 38 liters (10 gallons) of acetone spread out over an area of 4 square meters was postulated. With the parameters used in the computer code, this fire results in an intensity of 4.67 MW for approximately three minutes. The total heat released in the fire is approximately 8.9E+08 Joules. The volume of this spill was determined by examining several Appendix R submittals. Typically, a liquid spill is postulated in these analyses near safety-related equipment and on the order of 38 liters is required to cause a problem based on utility analyses. Postulating a 38 liter spill that is of the same relative size as that used by the utilities provides a means to verify results by using a different calculational tool. Additionally, tests performed at Sandia to evaluate the 20-foot cable separation requirement of Appendix R required an equivalent amount of a liquid fuel to cause problems [7]. A 2-meter square pool was chosen to represent an "in between" size pool. Although the actual size would tend to be larger for an unconfined pool (a depth of about 1.75 mm based on the surface tension of acetone), the resulting fire would be very intense and of very short duration, probably insufficient to cause cable damage. On the other hand, a smaller pool would burn less intensely, but for a longer period of time. (A smaller pool is partially considered using the trash fire which burns at 500 kW for 15 minutes.) Because it is not known exactly how and where a liquid spill would occur, these two fires were chosen in an attempt to keep the analysis conservative. Electrically-initiated fires are not being explicitly evaluated, but it is assumed that the exposure fires postulated for this analysis bound any transient or electrically-initiated fire that might occur at a plant.

In addition to the above fuel parameters, the worst case geometries are used in assessing fire growth and spread (e.g., most in situ fuel loading in an area, redundant cables pass the closest to one another, etc.). With these assumptions, the fire code COMPBRN was used to determine if the cables or components of interest would be damaged by the fire. This code has been used extensively by other fire PRAs and is designed to handle cable tray geometries. The code was used in a natural ventilation mode to evaluate the effects of the fire since its ability to handle a forced ventilation problem has not been adequately benchmarked to date.

d. St. Lucie Unit 1 transient event trees are used to determine the systems/components, from a success-oriented outlook, that need to be considered.

A success oriented approach was used since 1) if the "OK" states on the event tree cannot be negated, then a fire scenario cannot lead to a core melt (except with random failures considered), and 2) if it is shown that an "OK" state

0474

is vulnerable to a fire, the core melt probability has increased. By knowing what the vulnerable system(s) are, "modifications" can be recommended that significantly reduce the core melt probability. In this way, the "best" modification can be recommended rather than trying to fix everything. Transient event trees were used in this analysis since it was assumed that a fire occurs independently of any initiating event and the operators will always insure that at least a transient will occur once they know that there is a fire (i.e., scram the plant). This approach is consistent with previous PRAs. Fire's will not be assumed to start as a result of a transient. For LOCAs, the findings of previous PRAs have shown that, in general, the probabilities of LOCAs occurring as a result of or in combination with a fire are small. Therefore, those systems required solely to mitigate LOCAs will not be analyzed.

e. Components affected by a fire are electrical/active in nature (e.g., cables, motors, switchgear, buses).

Electrical equipment is by and large susceptible to fires since a fire either changes the electrical properties of the components or physically causes damage (e.g.,insulation damage, melting of contact points, shorts or opens). Although motors are not expected to be a critical failure component, they are assumed to fail in five minutes at 20% above their maximum manufacturer's specified operating temperatures. Sensitive electrical components and those electronic systems typically found in electrical cabinets are assumed to fail at an internal cabinet ambient temperature of 150°F (based on values used in previous PRAs and a review of commercially available integrated circuit manufacturers catalogs). Buses will fail at a similar temperature since fuses and breakers might fail rendering the bus inoperative. Cables will be assumed to fail at their autoignition temperature or from time at temperature based on tests run at Sandia for cabling "cooked" at a constant temperature until failed [8]. The performance of the particular cable types used in the St. Lucie Unit 1 plant may vary from the cable tested at Sandia. However, it is assumed, without test proof, that the variations are insignificant. The time at temperature failure criteria are given in Table D.4 for qualified and unqualified cable.

Only temperature will be used to evaluate component vulnerability to a fire. Passive components, such as pipes and manual valves will be assumed to be unaffected by a fire.

Note: Other potential damage mechanisms, such as smoke, corrosive vapors, increased humidity due to the fire environment or due to suppression activity (except in a bounding sense for manual suppression) are not being used to evaluate component damage. These effects could have significant impacts on the core melt probability because they may effect components directly in the fire zone or in other locations (i.e., smoke could be transported via a ventilation systems or by other means to another zone and cause inadvertent suppression system actuation or direct equipment damage. Thus a seemingly insignificant fire might cause significant damage.) There presently does not exist a data base that could be used to quantify these effects. In this analysis credit is given for automatic suppression systems actuating and not causing subsequent component damage. However, these automatic systems might in fact cause damage and the reader should keep in mind that this could only increase the core melt probabilities presented in the analysis section of this report. However the magnitude of this increase is unknown.

f. Random failures of those systems not directly affected by the fire at a particular location will be considered if applicable.

For those cases where a particular safety train is not affected by the fire or where a remote shutdown capability exists, random failures of those systems left after the fire will be evaluated. These would be the systems or trains that for the short term would be expected to provide decay heat removal. This approach ensures that credit is given for these systems but that the potential for their incorrect operation is factored into the analysis (i.e., after a fire, a system could still randomly fail).

g. Offsite power was assumed to be available during a fire unless the fire was capable of causing a loss-of-offsite power.

In most situations, it was assumed that given a fire and a reactor scram, offsite power would be available to the plant for recovery actions. This assumption only affects those sequences that include the power conversion system in one of the success paths on the event tree. Thus, all possible success paths that are indicated on the transient event trees were included in the analysis.

h. Fire growth and spread and suppression of fires are treated as independent of each other.

Consistent with other fire PRAs, this assumption was made to make the analysis a workable problem. In a "real" fire, as suppression is applied, the fire would not instantaneously be extinguished, but would slowly decrease in flame size and energy output. However, there are currently no models available that can adequately address these two competing processes. Consequently, when suppression action takes place, the fire was assumed to be instantaneously put out.

i. No explicit core melt timing was evaluated.

The TAP A-45 systems analysis includes thermal-hydraulic phenomenology in determining the timing for internal event sequences to determine potential recovery time windows. Typical time windows before core melt ensues are several hours. However, for the fire analysis, it was assumed that the repairs that would be required to be made after a fire had occurred could not be accomplished in these time windows because of the multitude of damaged cables and components that might be expected. Therefore, the core melt frequencies do not reflect heroic actions that might be taken by plant personnel to return some decay heat removal systems to operation prior to the onset of core melt. This assumption is a conservative approach and is consistent with previous fire PRA assumptions.

j. Where exact locations of cables cannot be accurately determined, assumptions as to their locations are made.

Because of the time and effort that would be involved in tracing the locations of cables in say, a cable spreading room, simplifying assumptions are made as to the locations of redundant cables unless the documentation readily available shows the positions of these cables. For instance in the cable spreading room, a horizontal separation of 3 ft and a vertical separation of 5 ft for redundant cables will be used if no other information is available.

k. Credit is given in the calculations for use of radiant energy shields, conduit, solid bottom cable trays, and completely enclosed cable trays.

Where radiant energy shields were used to prevent direct flame impingement, credit was given for the shields to the extent that only hot gas layer damage was considered where the cables would otherwise be in the plume directly above the flames. Also, rather than assuming that the cables autoignite at their piloted ignition temperature, they were assumed to ignite at their autoignition temperature. This latter assumption was also used if cables were in solid bottom cable trays. Where conduit or cable trays with solid tops and bottoms were used, the only credit allowed was that the cables would not contribute any fuel to the fire and that the cables would not be instantaneously damaged until they reached their autoignition temperature.

Note: The approach used in this analysis is designed to primarily look at the effects of a fire on a single unit. The results from the analysis will be presented for one unit, but if applicable, insights on how the fire could affect a two unit plant will also be included.

2. APPLICATION OF METHODOLOGY

The implementation of the methodology previously presented will be applied to the St. Lucie Unit 1 Nuclear Power Plant in this section. This section follows closely the steps previously presented.

2.1 Initial Screening to Determine Potentially Important Fire Areas for Further Analysis

Identification of Systems

This step involved the identification of those systems and their associated success criteria from transient event trees developed for the plant that lead to 'OK' states on the event trees. The transient event tree examined was T3, Transient with Power Conversion System Initially Available. This event tree best matches the assumptions used in this analysis as well as the fact that the other two transient event trees constructed for St. Lucie Unit 1 are just modified subsets of this event tree, and T3, as such, contains all the success paths being modeled. From this transient event tree, three success paths were identified. These success paths were made up of the following systems or combinations of systems as follows:

a. Power Conversion System (PCS) alone

b. Auxiliary Feedwater System (AFWS)

c. Safety Injection (SI) System in Feed and Bleed mode leading to High Pressure Recirculation (Safety Injection Pumps and Residual Heat Removal (RHR) pumps and Heat Exchangers available)

From this event tree, if any of these systems could function in a transient as designed, the core would not be damaged. Therefore, for this analysis, the three front line systems of importance became combinations of a, b and c above. (See systems analysis section for presentation of event trees.) The next step, once the front line systems had been identified, was to determine their success criteria. The success criteria for these systems was obtained from the systems analysts and is summarized below.

a. Power Conversion System: 1 out of 2 trains
b. Auxiliary Feedwater System: 1 out of 2 motor driven pumps or the turbine driven pump train
c. Safety Injection System: 1 out of 3 trains
d. Residual Heat Removal System (in recirculation mode): 1 out of 2 RHR pumps in series with 1 out of 2 RHR heat exchangers

For the front line systems the success criteria has an important implication for this analysis. Because in general the success criteria only require that 1 train or pump from each system is needed, to invalidate the success criteria implies that locations must be found in the plant where all trains of a particular system can be damaged by a fire. More importantly, this means in general that for this analysis, only those locations where all the trains of the systems listed above can be damaged are important. (Note: in general this is true, however locations may exist where a front line system may not have all its trains together, but its support systems may, therefore support systems must also be carefully examined.)

As per the methodology, not only front line systems were examined, but also the support systems required for the front line systems. The systems analysis section identifies the support systems necessary for continued operation of front line systems. In general the systems that are required are:

a. Intake Cooling Water (ICW): provides cooling to the following: component cooling water heat exchangers, turbine generator cooling systems

b. Component Cooling Water (CCW): provides cooling to the following: safety injection pumps, RHR pumps and heat exchangers, and other systems not considered as part of this analysis

c. AC/DC Power: power is provided to all the pumps that have been previously mentioned as well as actuation signals and controls for motor operated values

The success criteria for the support fluid systems are:

a. Intake Cooling Water: 1 out of 3 ICW pumps
b. Component Cooling Water: 1 of 3 CCW pumps with 1 out of 3 CCW heat exchangers

Once again, it can be seen that to invalidate a support system criteria, only a fire in those locations that have all of the trains in a system can effectively do this. This information is used in the next step to determine which plant areas need to have a more detailed analysis performed for them.

Screening to determine potentially important fire areas

The next step in the analysis was to determine which areas in the plant could meet the requirement of having enough of either front line or support systems to invalidate all the success paths possible for the plant. This was done by a review of readily available plant documentation and mapping the locations and components (refers to both components and cables) into a matrix, as a first screening, to determine if an area had enough of the combinations of systems to get one into trouble. For this analysis, the utilities latest Appendix R submittal [9] was used to determine the components in a given area.

Table D.5 is a matrix of the areas of interest and as a first screen determines if any of the front line or support systems identified by using the above documentation appear in the fire zone of interest. (Note: this table shows if 1 train is running through a location or if it has all the systems and trains. This matrix also includes instrumen- tation, control, and power lines for the safety systems being considered.)

From the table, several zones can be automatically eliminated since they do not contain any of the success oriented systems of interest. These are fire zones 9, 10, 11, 12, 18, 27, 29, 37, 39, 40, 41, 42, 43, 44A, 45, 46, 48, 49, 50, 51, 52, 53, 62, 64, 65, 66, 67, 68, 69, 71, 72, 73, 74, 79, and 81. Of the other zones that are left, some of these zones can also be eliminated by inspection based on the method being used in this analysis since not "all" of the success branches on the T3 event tree could be invalidated. Specifically, the following fire zones can be eliminated:

Fire Zone	Reason for no further analysis:
31 (Shutdown Heat Exchanger 1B Room)	Fire zones 31, 32, 33, 34, 35, 36A, 38, 44, 46, 75, 76, 81, and 82 can be eliminated from further analysis since the PCS and AFWS would be available given fires in these zones.
32 (Shutdown Heat Exchanger 1A Room)	
33 (Piping Penetration Room)	
34 (1A Emergency Core Cooling System Room)	
35 (1B Emergency Core Cooling System Room)	
36A (Charging Pump Access Hallway)	
38 (Charging Pump 1C Cubicle)	
44 (Radio Chemistry Lab)	
46 (Containment Purge Room)	
75 (Charging Pump 1B Cubicle)	
76 (Charging Pump 1A Cubicle)	
81 (Area Between Auxiliary Building and Containment)	
82 (Component Cooling Water Surge Tank Room)	

Fire Zone	Reason for no further analysis:
13 (Turbine Lube Oil Reservoir Area)	The AFWS and SI systems would be available given a fire in zones 13, 14, 15, 17, 19, 20, 21, 22, 23, 25, 63, and 80.
14 (Condensate Pump and Condenser Area)	
15 (Condensate Pump Pit)	
17 (Turbine Cooling Water Pumps and Heat Exchanger Area)	
19 (Feedwater Pumps 1A and 1B)	
20 (Heater Drain Pump Area)	
21 (Non-Class 1E "A" Switchgear Room)	
22 (Non-Class 1E "B" Switchgear Room)	
23 (L.P. Heater Area)	
25 (Turbine Deck)	
63 (Closed Blowdown Heat Exchanger Area)	
80 (Steam Generator Blowdown Tank Room)	
26 (Containment)	Fire zone 26 is not considered as part of this analysis.
3 (Intake Cooling Water Area)	One success path, AFWS would be capable of responding to transient events for fires in zones 3, 5, 8, 47, 61, 77, and 78.
5 (Component Coding Area)	
8 (Refueling Water Tank)	
47 (Class 1E 'AB' Switchgear Room)	
61 (HVAC Equipment Room)	
77 (East Electrical Penetration ('A' Train) Room)	
78 (West Electrical Penetration ('B' Train) Room)	
1 (Steam Trestle Area)	One success path, SI, would be capable of responding to a fire event in zone 1. In addition, this zone is a large semi-open area, thus hot gas layer effects would not be felt by components, or cables in this area.
2 (Condensate Storage Tank Enclosure)	Both the PCS and SI systems would be available given a fire in zone 2. (This assumes a fire does not effect tanks or values.)

The previous fire zones were eliminated from further consideration because by inspection of Table D.5, they did not contain "enough" of the safety systems of interest. However, other fire zones may also be eliminated from analysis based on a more detailed examination of exactly what systems are found in the fire zone of interest. The following fire zones from Table D.5 can be eliminated for the following reasons:

Fire Zone	Reason for no further analysis:
4 (Diesel Oil Storage Tank Area)	The loss of this zone could ultimately lead to the loss of both of Unit 1's emergency diesel generation (i.e., they would run out of fuel.) However, since a fire in this area would not affect offsite power, all success paths (PCS, AFWS, and SI) would be available to mitigate a transient.
6 (Diesel Generator 1A Room)	This zone contains Emergency Diesel Generator 1A (one of two safety-related diesel generators at St. Lucie Unit 1). A loss of this zone due to a fire would not affect offsite power, and therefore, all success paths (PCS, AFWS, SI) would be available. In addition, if offsite power were inadvertently lost, Emergency Diesel Generator 1B would be available and could be used to supply power to the AFWS and SI systems.
7 (Diesel Generator 1B Room	Similar rational as for fire zone 6, except Emergency Diesel Generator 1A would be available to supply necessary electrical loads.
16 (Main, Auxiliary and Start-Up Transformer Area)	A fire in this zone could affect the normal power to the PCS, AFWS, and SI systems that are of interest. However, Emergency Diesel Generators, 1A and 1B, could supply emergency power to the AFWS. Therefore, one success path would be available given a fire in this area.
36 (Main Hallway (-.5" elev.))	This fire zone only contains one train of safety-related equipment. Thus, its loss would still leave, as a minimum, one train of AFWS and SI systems available.

44A ('A' Cable Loft Enclosure)	This zone is physically located below the Class 1E 'A' switchgear room. As such, only A train safety-related cables run through this area. Train B PCS would be available given a fire in this zone since none of its cables run through this zone. In addition, B train AFWS and SI systems would be available.
55E (Main Hallway (elev 19.5') East)	This zone contains cables for train B safety equipment of interest. Thus, fire in this zone would cause the loss of train B equipment, but train A systems would be available to mitigate a transient event caused by a fire in this zone.
55W (Main Hallway (elev 19.5') West)	Although this zone contains cabling for a number of the safety systems of interest, it does not contain all the redundant trains for a particular system. For instance, power for AFWS pumps 1A and 1C run through this area (motor and steam driven AFW pumps respectively). However, power and control cables for AFWS pump 1B (motor driven) do not run through this area. Therefore one train of the AFWS success path would be available. In a similar manner, 1 train of SI would also be available given a fire in this area. In general, a fire in this zone would affect train A equipment only. Thus train B systems of PCS, AFWS, and SI could respond to the fire.
56 (Class 1E 'B' Switchgear Room	This zone distributes power to train B components in the PCS, AFWS, SI, RHR, CCW, and ICW systems. A fire in this zone would cause the loss of train B components; however, train A components (powered from zone 60) would be available in the PCS, AFW, and SI systems to mitigate a transient event caused by a fire.
58 (Battery 1B Room)	A redundant function is provided by station battery 1A, which does not have any cables running through this zone. Therefore, the PCS, AFWS, and SI system would be available to mitigate a transient.

59 (Battery 1A Room)	Similar to fire zone 58, however, station battery 1B provides the redundant function.
60 (Class 1E 'A' Switchgear Room)	Similar to fire zone 56, however, the redundant function is provided by the Class 1E 'B' Switchgear Room (zone 56), ensuring train B of PCS, AFW, and SI systems would be available.

Note: Because the areas above have not met the screening criteria being used for this analysis to determine critical areas, it is not, nor should it be implied, that they are not important from a fire standpoint. Quite the contrary, since the utility asked to have Appendix R exemptions granted for some of these zones. The key to remember for this analysis is that due to its limited scope, only those areas with truly a significant fire threat are being analyzed. The fire zones listed previously do not meet the criteria used in this analysis and that should be all that is inferred.

The two areas left must now be examined in more detail to determine if they have the requisite number of systems, that if they were all lost in this location, none of the system success criteria would be met. Therefore the next step is to look at each of these fire zone separately to determine if it needs to be further analyzed. Before proceeding, a brief discussion of the two areas is provided.

Fire Zone 57: Battery Charger, Inverter, and Cable Spreading Room

This zone contains all control and instrumentation cabling for the front line and support systems of concern. Also found in this zone are the reactor protection cabinets (It is assumed for TAP A-45 analyses that a reactor scram is a given, so that ATWS situations are not analyzed. This implies that for this analysis that even if a fire did effect these cabinets, they would fail safe). A fire in this zone would have the potential to cause damage to the control circuits of front line and support systems, thus affecting the reliability and operability of these systems.In addition, the possibility of maloperation of the power operated relief valves would also be possible given a fire in this zone. The capability to use a remote shutdown station is designed into the plant to allow for recovery (via the auxiliary feedwater system) from a fire in this zone.

Fire Zone 70: Control Room

This zone is very similar to the cable spreading room in that all control and instrumentation cabling for front line and support systems are found here. A fire in the control room could pose the same threat as the cable spreading room, but the capability to use a remote shutdown station would allow for recovery (via the auxiliary feedwater system). However, because this would require the evacuation of the control room, and human actions would determine if recovery were possible, this zone also should be analyzed in more detail.

The fire zones identified from the initial screening process requiring further detailed analysis were the following:

Fire Zone	Nomenclature
57	Battery Charger, Inverter, and Cable Spreading Room
70	Control Room

2.2 Quantification of Selected Fire Zones

This section quantifies the residual risk due to a fire in the two zones of interest. Note that in this quantification certain assumptions are used (as previously presented) that may terminate analysis steps. If it is shown, based on the assumptions being used, that no damage occurs to the components in a particular fire zone, then the analysis will not proceed and it will be assumed that the residual risk for such an area is less than 1.0E-6/Rx yr. As such, the cost/benefits (to prevent a potential core melt from a fire induced transient) would be minimal for any proposed modifications in this zone.

2.2.1 Fire Zone 57: Battery Charger, Inverter, and Cable Spreading Room

Description of the fire zone: This zone houses the Unit 1 reactor protection cabinets, battery chargers, inverters and distribution panels for DC buses 1A, 1B and 1AB, as well as other test equipment. The only other items found in this zone are the instrumentation and control circuit cabling for both safety-related and nonsafety-related systems in the plant. This includes the intake cooling, component cooling water, safety injection, RHR, auxiliary feedwater, and PCS systems that are of interest. The zone has a three hour fire rating (roof, floor, walls, doors, penetrations). Fire protection systems found in this zone include (a) photoelectric smoke detectors, (b) automatic, single failure proof, Halon suppression system, and (c) portable CO_2 fire extinguishers. In addition, a three-hour rated barrier has been built around the "A" battery charger, inverter, and DC distribution panel. Cables are routed into cable trays above this enclosure. Located outside this zone are manual hose stations. This zone is a large enclosure and as such, because of its function, has a large amount of cabling overhead (cable trays stacked 6 high, several stacks next to each other). Only a limited number of power cables enter this room.

1. Fire Occurrence Frequency for the Cable Spreading Room

The fire occurrence frequency for this room is taken directly from the generic fire frequency data base.

Cable Spreading Room Fire Frequency: 6.7E-03/Rx yr

2. Time to Damage Components in the Zone

The only burnable material in this zone is cable insulation. It is found in either steel cabinets, which would provide some measure of protection from an exposure fire, or the bulk of the cabling is found in horizontal cable trays. The cabling (PE/PVC type) is coated with a fire retardant Flamastic 71 or 77 coating. Newer cabling found in this zone is IEEE-383 qualified cabling. The cables of interest run in ladder type cable trays (the lowest cable tray in any stack has a solid bottom) and thus a fire could damage them either by direct flame impingement or by hot gas layer temperatures in the room being sufficiently hot and maintained long enough to cause the cables to fail due to internal melting. Cable tray heights range from approximately 7 feet off the floor and are found in stacks of several cable trays. In cable tray stacks, cable trays are separated vertically by approximately 1 foot. The separation horizontally between cable trays is also approximately 1 foot. To determine if this zone was susceptible to a fire, a COMPBRN simulation was run for this zone to determine what the predicted hot gas layer temperature was. (Note: Credit was given for the flamastic coated PE/PVC cabling as follows: It was assumed that the cables piloted ignition and spontaneous ignition temperatures were the same as IEEE-383 qualified cable. It was also assumed that the PE/PVC failure times at temperature were doubled because of the insulating capability of the flamastic. (This was based on a simple, one dimensional heat transfer model and the use of error functions to determine times to failure). Note that these assumptions (which have not been experimentally verified) are designed to give credit to flamastic fire retardant coatings, however the actual protection provided is unknown and these assumptions are not to be construed as endorsing fire retardant coatings. Two sets of cables were modeled, the first set containing five cable trays stacked with a 1 foot high vertical separation. The second set of cable trays was approximately three (3) feet away, and represents where a set of redundant cables might be found in the room. The results were as follows:

Trash Fire: The maximum hot gas layer temperature predicted by COMPBRN was not capable of causing damage to the cables.

Acetone spill - The cables directly above the fire ignited at about 240 seconds and began burning out at approximately 900 seconds. Redundant cables in the other cable trays were damaged by the hot gas layer. The time at which this was predicted to occur (i.e., when the cables were postulated to be irreparably damaged) was at approximately 900 seconds (15 minutes) into the fire. (A similar, although less conservative calculation for the Turkey Point Nuclear Power Plant predicted a damage time of 16 minutes.)

Therefore the time in which manual suppression could take place in this fire zone was taken as 15 minutes. This time was used in the next section to determine the suppression probability for manual suppression.

Note that because the situation modeled showed that there would be damage to redundant cables, for those cables not separated as widely, a fire in the given location would also be expected to damage them. Therefore the acetone fire in this area is predicted to damage all the cables of interest.(i.e., intake cooling water, auxiliary feedwater, safety injection, component cooling water, residual heat removal and power conversion system cabling).

3. Suppression Probability in the Zone

Two suppression probabilities needed to be determined for this zone. One probability concerns the automatic Halon system found in the zone while the other concerns the probability of manual suppression in the fire zone given that the automatic system was not successful or failed to function.

The probability of failure of an automatic Halon suppression system is .2 (see Table D.2).

The probability of the failure of a manual suppression team to suppress the fire prior to it causing damage (the time for this was determined in the previous step and was calculated to be 15 minutes) is approximately .3 (see Table D.3). No suppression system damage is considered.

4. Human Reliability Factors and Random Failures

As previously mentioned, a fire in the cable spreading room at St. Lucie Unit 1 could potentially cause the loss of all front line decay heat removal systems as well as their support systems. The fire would cause a loss of control of these systems which could cause them to fail in undesirable states. To overcome this situation, St. Lucie Unit 1 would have to rely on lining up its Hot Shutdown Panel (HSP) to provide cooling to the core. The location of this room is one level lower than the control room (where operators would be expected to come from to man the HSP). The HSP is located in fire zone 56. The HSP is designed to provide power to train A components, specifically, AFWS pump A would be used to remove decay heat from the core given a fire in the cable spreading room (references [10] and [11]). Thus to bring the HSP up on line so as to provide decay heat cooling would require plant personnel to become involved in manually lining up various systems or taking control to start these systems up (reference [10] delineates the responsibility of plant personnel in manning various stations to provide core cooling with the HSP). Because the fire that occurs has serious consequences and because there would, in all likelihood, be a highly confused state at the plant after the occurrence of a fire of this type, a human failure probability of .1 is assumed for lining up the HSP. (i.e. there is a 1 in 10 chance that the HSP and thus AFWS would not be lined up properly).

Note: A fire in the cable spreading room could have the potential to cause spurious operations to occur which would be highly undesirable. One of the events might include the loss of offsite power because of a spurious trip signal to the offsite power breakers. In this case, the station batteries would provide power to operate the AFWS and provide power to necessary instrumentation to control the plant for approximately eight hours before they would be depleted and in need of charging. After this time the plant status would be essentially unknown and therefore not controlled. However, because the cable spreading room is designed to protect the train A battery chargers, and A train DC distribution panels, the charging system for the batteries would be left intact. Offsite power could be restored to the charging system by clearing any associated trip signals in those breakers directly feeding the battery chargers. (This assumes the on site emergency diesel generators are not available.) This would obviously take time, and there would be a chance that it would not be done correctly, however, it is assumed that this failure probability is incorporated into the human reliability failure probability given above for lining up the HSP.

Similarly, given that the A train motor driven auxiliary feed water pump is lined up and placed in service properly to facilitate decay heat removal, there is still a chance that a random failure of the system could occur. The random failure probability for just the 1 train of motor driven auxiliary feedwater is approximately 1.0E-02/demand.

5. Final Fire Quantification for the Cable Spreading Room

The final overall fire risk contribution leading to core melt for this zone was performed by use of an event tree. Figure D-1 shows the fire event tree used in the quantification for this zone.

Overall core melt probability due: 4.38E-05/Rx yr
to a fire in this zone

Note that the dominant path in this analysis is that path that assumes all suppression system attempts at mitigation are

0488

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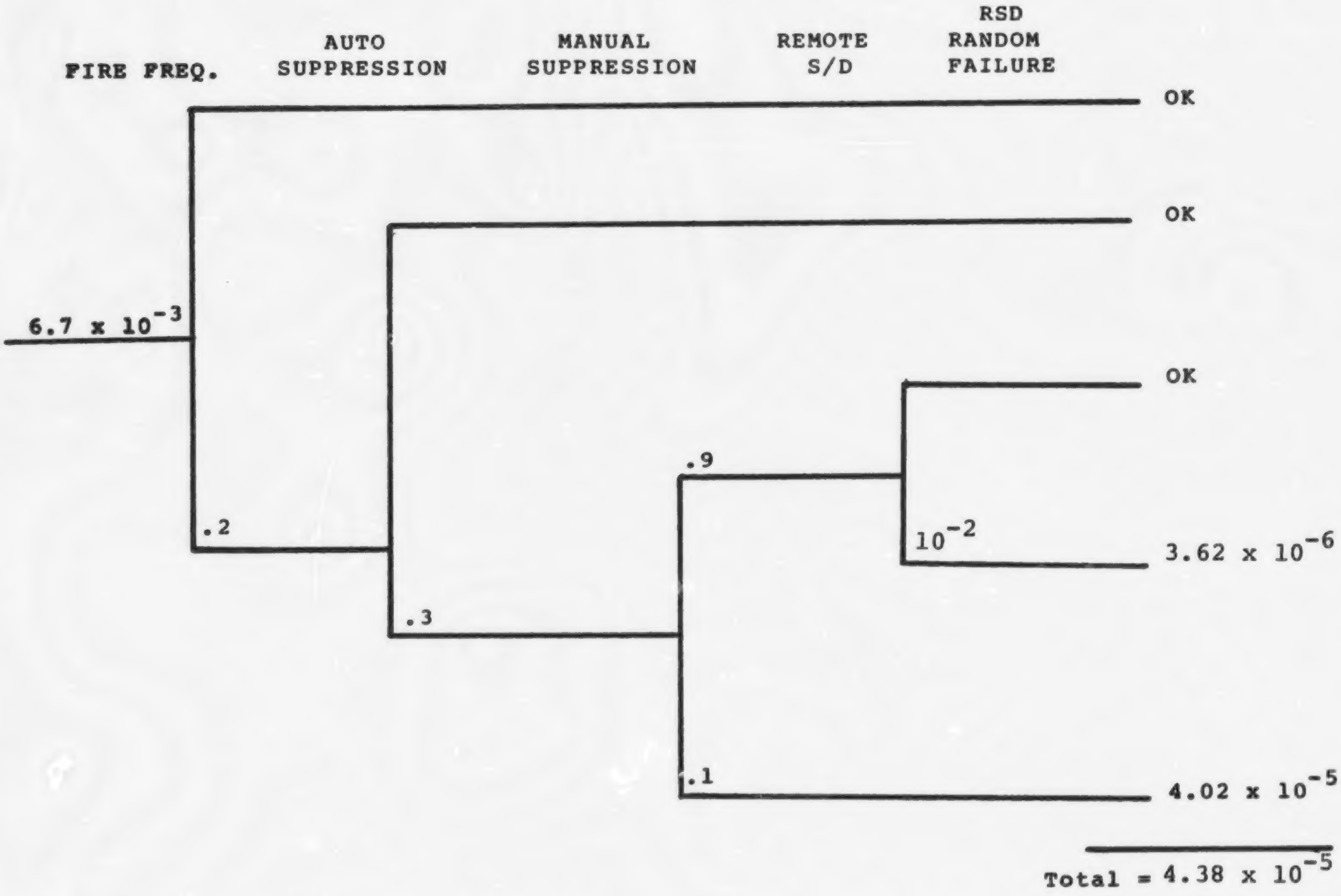
FIRE FREQ.
AUTO SUPPRESSION
MANUAL SUPPRESSION
REMOTE S/D
RSD RANDOM FAILURE
6.7 x 10^{-3}
.2
.3
.9
.1
10^{-2}
OK
OK
OK
3.62 x 10^{-6}
4.02 x 10^{-5}
Total = 4.38 x 10^{-5}

Figure D-1 Cable Spreading Room Fire Event Tree

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unsuccessful prior to damaging the cables of interest and that the actions required by plant personnel to properly line up the motor driven auxiliary feedwater pump are not done properly. Also, because this analysis does not attempt to look explicitly at the time that would be required to commence core melt given a transient, there may be situations where the human recovery factor could be reduced because of a larger time window in which the plant could be recovered. However, because the recovery times prior to core melt have been estimated by the systems analysts to range from less than 30 minutes to up to 2 hours, it is difficult to say if in fact this scenario were to occur that the human factors term could be reduced. (i.e., the exact sequence of events that this scenario leads to has not been analyzed and since it has been shown that damage occurs in 15 minutes, if one assumed that the plant was involved in fighting the fire during this time and was distracted from other operations, or in this case lining up the HSP for operation, the time window the plant would have to respond in at the maximum would be on the order of an hour and a half and might be on the order of only minutes).

6. Bounding calculation for the Cable Spreading Room considering Suppression Damage

How components are damaged by suppression actions is not a well understood phenomena, but is certainly a possibility that could occur for any fire due to the temperature stresses that a component would see from a fire and subsequent suppression activity, or due to a manual fire fighting team inadvertently directing a water stream onto previously undamaged components. The intent of this section is to simply look at the probabilistic bounds for the core melt probability developed for this area given that manual water suppression occurs and assuming that components not damaged by the fire are failed with a probability equal to 1.0.

In this zone, the damage that could potentially occur due to suppression damage could affect the control cabling for all the safety system success paths as well as some ESF DC distribution cabling. However, since the HSP is remotely located from this zone, and because manual suppression could only potentially accelerate the time in which redundant systems fail (i.e., the idea being that water sprayed on hot cables could potentially cause them to crack, leading to shorts), the core melt probability would be approximately three times as much as the value previously presented (assuming manual suppression has a failure probability of 1). This is true for this zone since all that is of interest is cabling, and manual water suppression in this zone could not damage large pieces of hardware (such as motors) because none are located in the zone. Therefore, an upper bound on the fire induced core melt probability for this zone would be 1.31E-04/Rx yr.

0490

7. Proposed Modification to Reduce the Core Melt Frequency due to a Fire in this Zone

The type of modifications that have been previously presented in the methodology section were considered for this zone. Of these modifications, only one appeared to be viable.

The proposed modification consists of wrapping all train 'B' cabling in a fire wrap to ensure that the train B safety systems can be used for some time before having to shift control to the HSP.

Cable Wrapping - The option selected. This is the simplest solution because it ensures that B train safety systems will be available for some time given a fire in the zone and because application of a thermal wrapping, once cables have been identified is a fairly straightforward task.

Construction of Barriers to isolate portions of the area - this would have entailed putting in a full height wall into the area. The number of penetrations that would be required through this wall would have been large and would potentially require rerouting of cables to ensure redundant systems were on opposite sides of the barrier.

Rerouting of affected cables - this option was not a viable solution since a large amount of engineering work would be required to determine the new routing and still meet Appendix R requirements. In addition, the actual rerouting process would be difficult and time consuming to perform and would have to be done while the plant was shutdown, thus potentially affecting its capacity factor if the work took longer than expected.

Automation of manual systems - not applicable for this zone.

Move components out of the area - only cables were of interest in this zone, therefore this option is similar to the rerouting of cables option which did not appear to be a viable solution.

Procedural changes - a fire in this zone potentially affects a large number of safety systems by causing electrical failure of these systems. Procedural changes would have little impact on mitigating the potential damage.

Install an additional suppression system - For independence, a water system would be required. However, at St. Lucie Unit 1, no automatic water systems have been used at any location in the plant. Installation of such a system would therefore be prohibitively expensive and the cost benefit derived would consequently be low.

So for this zone the best modification appeared to be wrapping B train cables.

The modified core melt probability obtained with this modification is determined as before.

By wrapping the 'B' train cables with a fire retardant material, their survivability is enhanced. The fire retardant wrap to be used is rated as a one-hour fire barrier (the fire wrap might provide protection in excess of an hour, which would further reduce the core melt frequency; however its survival is not documented past an hour; so for conservatism, this value is used). Therefore, the plant has essentially an hour to put the fire out before damage occurs. A fire lasting this long in the cable spreading room without being brought under control by either the automatic Halon system or from the plants fire fighting brigade is probably an incredible event. However, any fire lasting less than an hour, because of the close proximity of control cables in the cable spreading room, would necessitate the use of the HSP almost immediately without a fire wrap around train B cable. It is therefore assumed that the time for manual suppression (as an upper bound) in this room is one hour. The probability that the fire would not be manually suppressed in this time is .1 (Table D.3). (It is assumed that the combination of wrapping the 'B' train cables as well as the extra time this provides for manual suppression ensures that the HSP can be used to control the plant.)

As previously discussed, because the fire that occurs in the cable spreading room has serious consequences, and because there would in all likelihood be a highly confused state at the plant after the occurrence of a fire of this type, a human failure probability of .1 is assumed for properly lining up the HSP.

In addition, given that the motor driven auxiliary feedwater pump is lined up and placed in service properly to facilitate decay heat removal, there is still a chance that a random failure of the system could occur. A nominal random failure probability for just the one train of motor driven auxiliary feedwater is approximately 1.0E-02/Demand.

The modified fire event tree reflecting the modification is shown in Figure D-2.

Modified core melt probability with a fire retardant wrap around the 'B' train cables:	1.46E-05/Rx yr.

(Note: It should be kept in mind that the benefit derived from manual (water) suppression is predicated on the assumption that suppression itself does not degrade equipment operability.)

0492

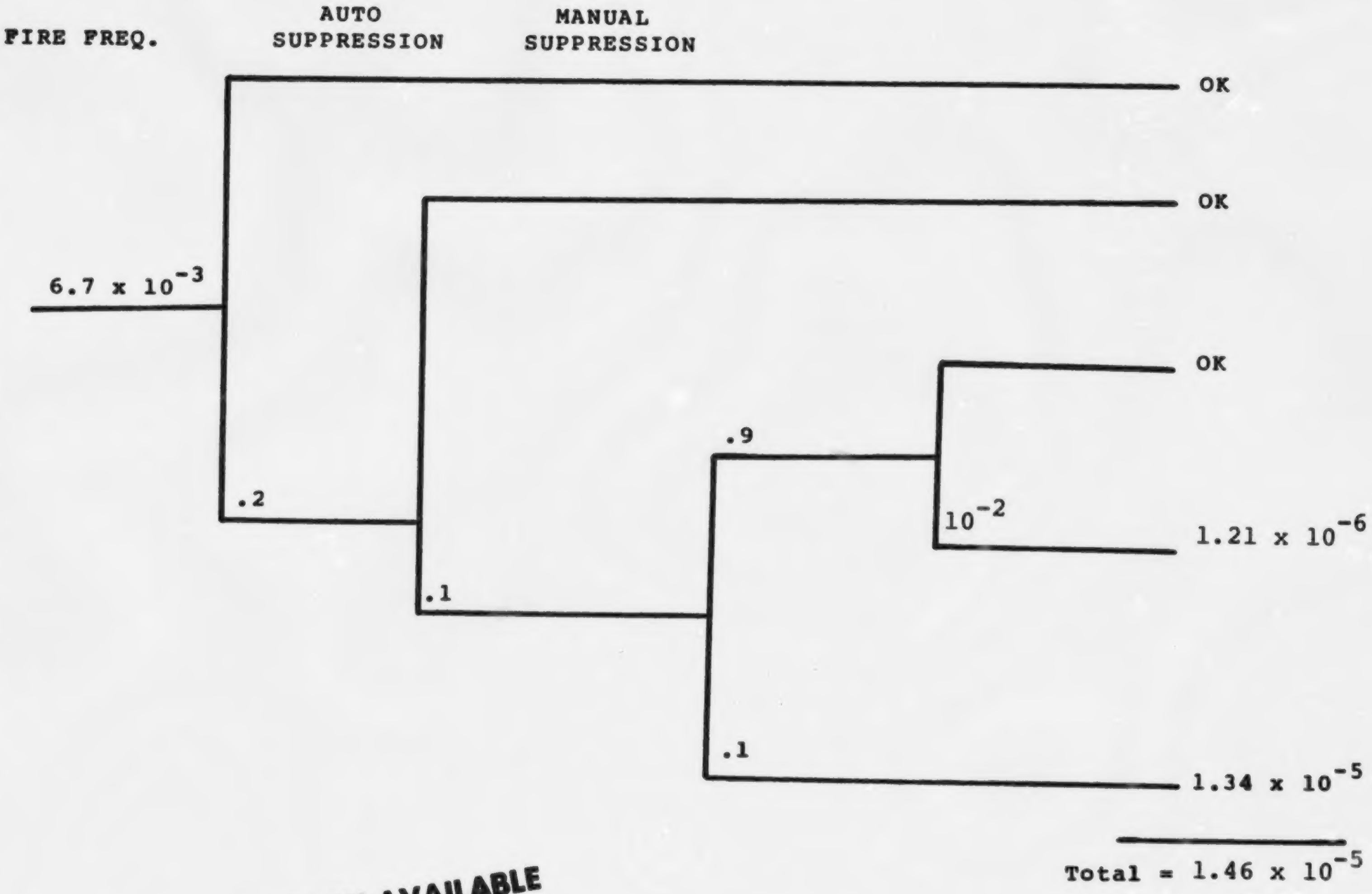
FIRE FREQ.
AUTO SUPPRESSION
MANUAL SUPPRESSION
6.7 x 10-3
.2
.1
.9
.1
10-2
OK
OK
OK
1.21 x 10-6
1.34 x 10-5
Total = 1.46 x 10-5

Figure D-2 Cable Spreading Room Fire Event Tree After Proposed Modification

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0493

2.2.2 Fire Zone 70: Control Room

Description of the fire zone: This zone contains instrumentation and control circuitry for all redundant safe shut down systems and other operating systems such as the main, power producing turbine generator, electrical distribution grid control, etc. Also found in this zone are test and calibration circuits. The bulk of the combustibles in this zone are due to either cabling or transient combustibles. Cabling is found inside steel cabinets and in general enters and exits the control room via sealed floor penetrations. This zone has a three hour fire rating. Fire protection systems for this zone include (a) photoelectric smoke detectors, (b) manual fire extinguishers and (c) a dedicated smoke removal system. Located outside this zone are manual hose stations. In this zone, the safety injection, turbine driven auxiliary feedwater pumps, intake cooling water pumps and component cooling water pump controls are found on different control board sections.

1. Fire Occurrence Frequency for the Control Room

The generic fire occurrence frequency for the control room is used.

Fire occurrence frequency for the Control Room: 4.9E-03/Rx yr

2. Time to Damage of Components in the Zone

In this zone, the systems that are of interest have controls on various control room panels that are separated from each other. The separation between the panels is on the order of 20 linear feet. This means that for the fire to cause damage to the systems listed above would require the fire to spread a minimum of this distance. In addition, because the cabling of interest only comes up about a meter into the room (to the sloped portion of control boards or about waist high in vertical cabinets), hot gas layer affects would not be expected to play a major part in causing damage. Only the direct impingement of a flame or heat radiation from a fire would cause damage. Because safety system cables of interest are not in the direct line of sight of each other (i.e., cables can not "see" the other safety-related cables due to obstructions from cabinet walls and supports), the time for flame or radiation damage to occur would be, in all likelihood, long. In addition, because this zone is continually manned, the chance of a fire spreading to a size where it could cause this type of damage is deemed to be small. Because not all of the systems that could be used to provide decay heat removal are located on the same panel, and consistent with the assumptions in this analysis and the rational just presented, further analysis of this zone is not warranted.

0494

*In considering CR fires, only obvious fire vulnerable cabinets were considered (i.e. those containing front line safety systems). Subtle interactions, such as spurious operation of components, effects of ventilation, and damaging effects which suppression might cause, were ignored, consistent with the original set of assumptions.

3. CONCLUSIONS AND INSIGHTS

At St. Lucie Unit 1 it appears that only one fire zone needs improvement to the fire protection features already employed to reduce the risk of fire in that zone. In summary, the fire zone, and the core melt frequency due to a fire, both before and after the proposed modification, is presented below.

Fire Zone	CM Frequency (before modification)	CM Frequency (after modification)
57 (Cable Spreading Room)	4.38E-05/Rx yr	1.46E-05/Rx yr

Proposed Modification: Wrap 'B' train cables with a one-hour rated fire retardant material. The wrapping will ensure one train of several safety systems are available for at least one hour given a fire in this zone.

Insights: Specific plant and uncertainty insights are presented below:

Plant Insights:

a. The quantification for the Cable spreading Room is in general applicable to both Unit 1 and 2 (i.e., the plants have similar designs, the principal difference being that Unit 2 uses a water suppression system and not a Halon suppression system like Unit 1 in the cable spreading room).

b. For the fire scenario postulated, only Unit 1 would be affected by the fire. The postulated scenarios limits itself to affecting only one unit.

c. The control room, which was not quantified, but was identified initially as having the potential to lead to a core melt situation, would have the capability to affect both units simultaneously given a fire of sufficient size and duration.

d. Other zones which were not evaluated because they did not contain all the systems needed to negate the success paths, if lost, could cause the loss of all trains of one safety system for Unit 1. These zones contained all of a particular safety system component(s) in one room. For example, the AFW Pumps are all located in one area. There are redundant safety systems provided given the loss of these systems; therefore, decay heat removal could be maintained. However, improved separation of redundant trains in newer plants points out the changes seen in design philosophy from an older plant such as St. Lucie Unit 1. (i.e. as compared to newer plants built with BTP 9.5-1 or Appendix R requirements in effect.)

e. At St. Lucie Unit 1, the pumps and motors for the front line decay heat removal systems and their supporting systems considered in this analysis, are physically widely separated at the plant. For instance, the AFWS pumps are located several hundred feet from the SI and RHR pumps. Because all these systems must be controlled from the control room, the only common points where these systems interface, at least electrically, is the control building complex. Both this analysis and St. Lucie Unit 1's own analyses (Appendix R) point this fact out.

f. St. Lucie Unit 1, Unit 1, was purposely designed by the utility to not have any automatic water suppression systems in any fire zone. Because of this, the installation of an automatic water suppression system would be prohibitively expensive.

Uncertainty Insights

Some of the potential areas of uncertainty are identified as follows:

a. Effects due to suppression activities (both automatic and manual) on safety equipment. This includes suppression effects in the zone where the fire is as well as the effects in other areas.

b. Effects due to smoke, corrosive gases and humidity changes caused by a fire on safety-related equipment.

c. The method of partitioning used to determine generic room or building fire frequencies from the fire occurrence data base. Similarly, the applicability of using generic fire frequency data representing all plants for an individual plant.

d. Manual suppression probabilities relating the time available prior to component damage to the probability of not being able to suppress the fire in the given time frame.

0496

e. Capability of deterministic codes to accurately predict the actual fire environment seen by safety equipment when a fire occurs.

f. Electrical independence between control room circuits and remote shutdown type panels (either one central panel or several small panels located in different zones) that allow for control of safety equipment if the control room is evacuated.

g. The actual fire protection provided by fire retardant coatings, both due to direct flame impingement and when continually exposed to hot gas layer effects and temperatures.

h. Human reliability probabilities relating to the change of maloperation of auxiliary shutdown panel components given a fire requiring their use.

0497

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5. TABLES

Table D.1

Fire Occurrence Frequencies for Various Nuclear Power Plant Locations [3]

Location	Mean Fire Frequency (per Rx-yr)
Control Room	4.9 E-03
Cable Spreading Room	6.7 E-03
Auxiliary Building	4.8 E-02
Turbine Building	1.6 E-02

Table D.2

Automatic Fire Suppression System Non-Reliability [2]

Type of Suppression System	Probability of Failure (per demand)
Halon	0.2
Carbon Dioxide (CO_2)	0.1
Water (both for wet and dry pipe systems)	0.04

0499

Table D.3

Manual Nonsuppression Probabilities as a Function of the Time Before Component Damage Occurs [3]

Time Available Prior to Damage	Probability of Non-Suppression
5 min	0.4
15 min	0.3
30 min	0.2
60 min	0.1

(Note: These probabilities imply that for manual suppression activities if, for instance, a fire occurs that damages components of interest in 5 min, the probability that it will not be suppressed within the 5 min is 0.4 or conversely there is a 60% chance it will be suppressed in this time interval.)

Table D.4

Cable Failure Criteria [6]
(Time at Temperature)

	Nominal Time to Electrical Failure (minutes)	
Cable Temperature (°C/°K)	Unqualified Cable (e.g. PE/PVC)	IEEE 383 Qualified (e.g. XPE/XPE)
250/523	7	-
350/623	4	13
450/723	2	4
550/823	-	3

Table D.5

Zone Matrix

	Fire Zone	PCS	AFWS	SI	RHR	ICW	CCW	PORV	AC/DC
1	(Steam Trestle Area)	X	X						
2	(Condensate Storage Tank Enclosure)		X						
3	(Intake Cooling Water Area)					X			
4	(Diesel Oil Storage Tank Area)								X
5	(Component Cooling Area)					X	X		X
6	(Diesel Generator 1A Room)								X
7	(Diesel Generator 1B Room)								X
8	(Refueling Water Tank)			X		X			
9	(Primary Water Storage Tank Area)								
10	(Gas Storage Building)								
11	(Miscellaneous Oil Storage Building)								
12	(Water Plant)								
13	(Turbine Lube Oil Reservoir Area)	X							
14	(Condensate Pump and Condenser Area)	X							
15	(Condensate Pump Pit)	X							
16	(Main, Auxiliary and Startup Transformer Area)	X	X			X			X
17	(Turbine Cooling Water Pumps and Heat Exchanger Area)	X							
18	(Instrument and Station Compressors Area)								
19	(Feedwater Pumps 1A and 1B)	X							
20	(Heater Drain Pump Area)	X							
21	(Non-Class 1E 'A' Switchgear Room)	X							
22	(Non-Class 1E 'B' Switchgear Room)	X							
23	(L.P. Heater Area)	X							

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TABLE D.5 (Continued)

Zone Matrix

Fire Zone	PCS	AFWS	SI	RHR	ICW	CCW	PORV	AC/DC
24 (Isolated Phase Bus Area)	X							
25 (Turbine Deck)	X							
26 (Containment)			X				X	
27 (Aerated Waste Storage Tank Area)								
29 (Chemical Drain Tank Area)								
31 (Shutdown Heat Exchanger 1B Room)				X				
32 (Shutdown Heat Exchanger 1A Room)				X				
33 (Piping Penetration Room)				X				
34 (1A Emergency Core Cooling System Room)			X	X				
35 (1B Emergency Core Cooling System Room)			X	X				
36 (Main Hallway (-0.5' elev))		X	X	X	X	X		X
36A (Charging Pump Access Hallway)			X					
37 (Boric Acid Condensate Pump)								
38 (Charging Pump 1C Cubicle)			X					
39 (Gas Decay Tank Area)								
40 (Boric Acid Make-up Tank Room)								
41 (Hold-up Tank Enclosure)								
42 (Chemistry Personnel Building)								
43 (Control Building Personnel Area)								
44 (Radio Chemistry Lab)				X				
44A ('A' Cable Loft Enclosure)		X	X	X	X	X		X
45 (Piping Penetration Room)								
46 (Containment Purge Room)								
47 (Class 1E 'AB' Switchgear Room)			X		X	X		X
48 (Letdown Heater Exchange Room)								

0502

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TABLE D.5 (Continued)

Zone Matrix

Fire Zone	PCS	AFWS	SI	RHR	ICW	CCW	PORV	AC/DC
49 (Volume Control Tank Room)								
50 (Demineralizer Area)								
51 (Drumming Station)								
52 (Boric Acid Water Concentrator Cubicles)								
53 (Laundry and Decontamination Area)								
55E (Main Hallway (elev. 19.5) East)	X	X	X	X	X	X		X
55W (Main Hallway (elev. 19.5) West)	X	X	X	X	X	X	X	X
56 (Class 1E 'B' Switchgear Room)	X	X	X	X	X	X	X	X
57 (Battery Charger Inverter & Cable Spreading Room)	X	X	X	X	X	X	X	X
58 (Battery 1B Room)								X
59 (Battery 1A Room)								X
60 (Class 1E 'A' Switchgear Room)	X	X	X	X	X	X	X	X
61 (HVAC Equipment Room)					X			
62 (Resin Addition Tank Area)					X			
63 (Closed Blowdown Heat Exchanger Area)	X							
64 (Fuel Pool Pumps and Heat Exchanger Area)								
65 (New Fuel Container Storage Area)								
66 (Spent Fuel Pool Area)								
67 (Cask Wash down Area)								
68 (New Fuel Storage Area)								
69 (Spent Fuel Pool HVAC Area)								
70 (Control Room)	X	X	X	X	X	X	X	X
71 (Control Room HVAC Area)								
72 (Auxiliary Building Roof)								
73 (Technical Support Center and Offices)								
74 (Spent Resin Tank Room)								
75 (Charging Pump 1B Cubicles)								
76 (Charging Pump 1A Cubicle)			X					

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TABLE D.5 (Continued)

Zone Matrix

Fire Zone	PCS	AFWS	SI	RHR	ICW	CCW	PORV	AC/DC
77 (East Electrical Penetration ('A' Train Room)			X			X		X
78 (West Electrical Penetration ('B' Train Room)	X		X				X	X
79 (Raw Water Storage Tank Fire Pump Area)								
80 (Steam Generator Blowdown Tank Room)	X							
81 (Area Between Auxiliary Building and Containment)								
82 (Component Cooling Water Surge Tank Room)						X		

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0504

APPENDIX E

INTERNAL FLOOD RISK ANALYSIS FOR THE ST. LUCIE NUCLEAR POWER PLANT

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TABLE OF CONTENTS

1. ANALYSIS METHODS AND ASSUMPTIONS 3 507

1.1 Introduction 3 507

1.2 Scope 3 507

1.3 Analysis Methodology 4 508

1.3.1 Screening for Potentially Important Internal Flood Areas 4 508
1.3.2 Plant Visit 5 509
1.3.3 Quantification 6 510
1.3.4 Proposed Modifications 7 511

1.4 Assumptions 8 512

2. APPLICATION OF METHODOLOGY 12 516

2.1 Initial Screening to Determine Potentially Important Internal Flood Areas 13 517
2.2 Quantification 16 520
2.2.1 Tank Induced Flooding 17 521
2.2.2 Flood Sources found outside the zones of interest 19 523
2.2.3 Largest flood source in each area 20 524

3. CONCLUSIONS AND INSIGHTS 20 524

4. REFERENCES 22 526

5. TABLES 23 527

1. ANALYSIS METHODS AND ASSUMPTIONS

1.1 Introduction

Operating experience at nuclear power plants has shown that the potential for damage to safety-related equipment as a result of internal flooding is of concern. Examples of how plants have experienced internal flooding include operators overfilling water tanks, in-leakage of ground water, hose and weld ruptures, pump seal leaks, improper maintenance procedures, and various other circumstances. Several of these events resulted in damage to high pressure coolant injection systems, reactor building spray pumps, service water pumps, emergency feedwater pumps, diesel generator control cabinets and other redundant and diverse safetyrelated components. Although to date, none of these floodings have led to a serious accident, there is still a risk that needs to be evaluated to determine the probability that an internal flood, despite a conscious effort to design the plant to withstand this threat, could lead to a core melt scenario. The TAP A-45 program specifically addresses these concerns for the decay heat removal systems and the sections that follow will present the methodology and assumptions employed to carry out the internal flood analysis.

1.2 Scope

a. This analysis will evaluate for the St. Lucie nuclear power plant the risk from internal flooding that could affect decay heat removal systems. The results of this analysis, the quantified risk and any modifications that may be proposed, will be compared to risk from other initiating events with the modifications evaluated for risk reduction and cost. This "value-impact" analysis will then ultimately determine the optimum set of modifications to get the desired reduction in risk.

b. The analysis is not intended to be a comprehensive internal flood PRA. An analysis of this sort would be beyond the scope of TAP A-45. The intent of this analysis is to quantify significant internal flood threats to a plant. In doing so, existing procedures used in published PRAs will be employed, but simplifying assumptions will be made to allow the analysis to proceed without requiring the level of detail required for these other PRAs. In that respect, this analysis embodies the basic approaches used by previous internal flood PRA's but is by necessity a scaled down version of these PRA's.

c. Wherever possible, data from previous PRAs will be used. Since this analysis embodies what has been done in previous PRAs, use of this data will provide consistency in how the data is used and allow some comparison

0507

between the analyses. The data is based primarily on that found in the Seabrook internal flood PRA [1].

d. Because this analysis is a scaled down version of an internal flood PRA, the quantification of the risk due to an internal flood will be presented as a point estimate. Uncertainty values will not be formally determined. However, insights into the potential major areas of uncertainty will be identified.

e. The systems to be examined and quantified are based on the event and fault trees developed by the systems analysts, and is thus constrained by the level of detail of the systems analysis effort. In a similar fashion, readily available plant documentation determines the level of physical detail used in this analysis. Documentation obtained for this effort is treated as accurate and up to date unless otherwise identified.

f. A screening cutoff of 1.0E-06 for an internal-flood-induced core melt probability is used in the analysis to determine if further quantification efforts in an area are warranted. This is based on insights from previous PRA's where the internal events analysis lead to core melt probabilities on the order of 1.0E-04. Any internal flood produced core melt probabilities less than 1.0E-06 would therefore be insignificant compared to the internal events.

1.3 Analysis Methodology

The approach used in determining the risk due to internal flooding at the St. Lucie nuclear power plant used the following four basic steps:

a. Initial screening for potentially important internal flood levels

b. Plant visit for verification

c. Quantification

d. Proposed modifications to reduce the risk due to internal flood in affected areas

1.3.1 Screening for Potentially Important Internal Flood Areas

The purpose of this step was to determine which levels in the plant have "enough" of the front line or support systems in their boundaries such that internal flood damage could invalidate all the success branches on the transient event trees.

The sequence of steps taken during this first screening were as follows:

a. Internal event analysis data was obtained as the starting point. The specific data included event trees, fault trees, and simplified piping diagrams used in constructing the fault trees.

b. The transient event trees were examined to determine those front line systems or combinations of systems that would lead to an 'OK' state on the event tree. Also obtained from the event trees were the success criteria of the front line systems.

c. The next step was to identify the support systems required for the front line systems identified above. Support systems included cooling systems and AC/DC power requirements. The success criteria for the support systems were obtained from the systems analysis. For all systems except the power conversion system, the fault trees for the front line systems were used to determine the support systems. For the power conversion system, plant schematics and piping diagrams were used to determine required support systems.

d. Once the systems of interest were identified, all available documentation (principally FSARs and plant supplied information and drawings) were analyzed to determine those areas where "enough" components could be disabled by an internal flood. Based on this screening, a small set of areas were left for further analysis.

To this point in the analysis, everything was based on a review of available documentation.

1.3.2. Plant Visit

A visit was made to the plant with the following objectives:

a. Based on the initial screening, the general location of components of the systems of interest are known. The next step was to visit the plant and see the physical arrangements in each of these areas. The plant visit also provides a means to determine locations of major flood sources and some estimate of the x, y, z coordinates of components. These locations are necessary for the quantification step.

b. The second purpose of the plant visit was to confirm with plant personnel that the documentation that was being used was in fact the best available information and to obtain clarification about any questions that may have arisen in a review of the documentation.

1.3.3 Quantification

For those areas remaining after the initial screening, the next step was to determine if in fact a flood source could damage the components of interest in the area, and if so, how much time would be available for operator action before all "OK" states were negated on the event trees. This calculation was based on Standard Review Plan 3.6.1 and 3.6.2 for determining break sizes for moderate energy lines. With the flow rate, room volume, and capability of water removal known, the time to damage of components was calculated based on the water depth in the room and the assumptions governing component damage thresholds. Spray was considered when a flood initiates in a room and the critical components are in a direct line of sight with the flood source. It was assumed that these components failed due to the spray; other critical components that were either in the same room or in different rooms could fail due to submergence caused by the flood source. Based on the results of the analysis, further quantification was either continued or the room was determined not to be susceptible to internal flooding (subject to the assumptions and limitations of the analysis).

An event tree was utilized to estimate the risk of core melt from internal flood/spray in those areas identified in the initial screening and not eliminated by subsequent analysis. The basic elements of the event tree are as follows:

a. Internal flood occurrence frequency for the area of interest.

b. Isolation potential of the flood source.

c. Operator error and random failures of unaffected systems.

Each of these elements are discussed below:

a. Internal flood occurrence frequency for the area of interest.

The internal flood frequency was based on operating experience data and is related to the size of the flood source. The flood frequencies used in this analysis are given in Table E1. [2]

b. Isolation potential of the flood source

Source Isolable from the Control Room - This step assigns a probability that the operator will correctly isolate the flood from the control room. If the flood is not isolable from the control room, this probability is set to zero. The probability is determined by using the time to submergence of the critical components and determining from human factors considerations the probability that the operator takes the proper isolation actions.

Note: Responses on control room instrumentation that alert the operators to a potential flood problem were not considered since a) it is not clear how these instruments would respond and display their information and b) the complex sequence of events that might take place would require detailed system and human reliability modeling efforts, which is beyond the scope of TAP A-45.

Isolable from the Plant - This step is similar to the above step, except that the human factors probabilities will be modified to reflect a more stressful situation since the operators would be assumed to be manually repositioning valves in the plant.

c. Operator error and random failures

An evaluation is made of operator error and random failures for any systems that are left unaffected by the internal flood. Specifically, any systems that could be used to provide a decay heat removal function are evaluated. If manual operator action is required at a remote location, the probability of operator error is taken as 0.1. If a system can be operated from the control room, the probability of operator error is taken as 0.05. If the system is fully automatic, the random failure of a single train is taken as 1.0E-02.

1.3.4 Proposed Modifications

Based on the quantification steps, engineering modifications were proposed for those areas which have an internal-flood-induced core melt probability contribution of greater than 1.0E-06/Rx yr. Proposed modifications are of the following nature:

a. Installation of automatic flood detection systems.

b. Construction of flood barriers to isolate portions of an area.

c. Physically moving redundant system hardware to other locations to prevent its loss during an internal flood.

For any proposed modification, a core melt probability was recalculated to indicate how much the core melt probability could be reduced. The quantification was performed by modifying the internal flood event tree to reflect the modification

The proposed modifications were supplied to an architectural engineering firm for cost evaluation during a second plant visit. A "value-impact" analysis which considered the core melt contributions from internal floods and other external and internal events was used to determine the "best modification" in an overall sense.

1.4 Assumptions

Most of these assumptions are based on insights derived from previous PRA's or by consideration of the physical constraints of a flood in a building. The major assumptions are as follows:

a. Internal flood frequencies, component damage thresholds, and physical room considerations will be based on previous industry wide data bases.

Existing data is used to the extent possible to evaluate the threat of an internal flood to a plant. Internal flood frequencies have been determined based on operating experience. These frequencies have been determined by the amount and flow rate from a flood source. For instance the Seabrook PRA has defined four levels of flooding:

1. Small - flooding on the order of hundreds of gallons, e.g. valve pit flooding, instrument flooding, or flooding within a component.

2. Moderate - flooding on the order of several thousand gallons, e.g.; a few feet of water on the floor of a typical pump room.

3. Large - flooding on the order of tens of thousands of gallons; e.g; a few feet of water in large rooms, very deep water (more than ten feet) in a typical pump room.

4. Very large - flooding on the order of hundreds of thousands of gallons, e.g. floods involving circulating or service water piping.

Other attempts at quantifying internal flood frequencies have considered the flow rate from the flood source.

The drainage available is based on an equivalent diameter approach (i.e. a square, 1 ft by 1 ft opening would be considered capable of handling the flow from a pipe with a diameter of approximately 1 foot).

Previous internal flood PRA's have used a factor of 20% to derate the floodable volume of a room. This accounts for equipment taking up some of the area. This analysis will use the same value.

b. St. Lucie transient event trees are used to determine the systems/components, from a success-oriented outlook, that need to be considered.

A success oriented approach was used since 1) if the "OK" states on the event tree cannot be negated, then an internal

0512

flood cannot lead to a core melt (except with random failures considered), and 2) if it is shown that an "OK" state is vulnerable to an internal flood, the core melt probability has increased. By knowing what systems are vulnerable, "modifications" can be recommended that significantly reduce the core melt probability. In this way, the "best" modification can be recommended rather than trying to modify everything. Transient event trees are used in this analysis since it is assumed that an internal flood occurs independently of any initiating event and the operators will always insure that at least a transient will occur once they know that there is an internal flood (i.e., scram the reactor). This approach is consistent with previous PRAs. Internal floods will not be assumed to start as a result of a transient. LOCA event trees will not be examined since pipe ruptures have already been considered in these trees. In addition, the findings of previous PRAs have shown that, in general, the probabilities of LOCAs occurring as a result of or in combination with internal flooding are small. Therefore, those systems required solely to mitigate LOCAs will not be analyzed. As an extension of this assumption, LOCA analyses will be assumed to bound all internal flooding which could occur inside primary containments and this analysis will exclude primary containment flood sources.

c. Front line mitigation systems (such as high and low pressure injection) will not be considered as flood sources.

The rationale behind this assumption is that front line systems a) have been designed to higher standards (seismically qualified, extensive material and workmanship quality assurance standards), b) have controlled chemistry, c) do not see large temperature gradients in general and d) have been involved in few catastrophic failures, and e) are not normally operating, therefore the chance of these systems being a flood source is minimal. Further, most front line systems only have a limited volume of water from which to draw suction. Typically this source is a tank. Because this analysis considers potential flooding in each zone from tanks emptying their contents into a single room, this is in essence no different than considering that a front line system pipe ruptures and runs until it completely empties its source (a tank). Therefore the tank analysis effectively bounds potential internal flooding from front line systems.

High energy fluid systems (i.e., greater than 200°F and 275 psig) will not be considered because of the following probabilistic and physical considerations:

1. The break of a high energy line, while it might cause damage to those components in its immediate vicinity, would not be able to damage all safety systems of interest due to spray since plants in general are designed such that

0513

redundant safety systems are not in the same physical locations. Therefore, the probability of core melt would be the probability of the break occurring times the random failure of the systems left. Typical numbers for these values respectively are 1.0E-2 and 1.0E-5. Thus the core melt probability would be on the order of 1.0E-7 which is below the cutoff being used in this analysis.

2. Plants are required to verify that the designs of components are sufficient to withstand the most severe environment that they could be expected to see. Therefore the environmental conditions that could be expected from a high energy line break are assumed to not affect components. (Note that the above is assumed to be true in this analysis, however, as pointed out elsewhere in this analysis, this may not be true.)

d. Flood barriers maintain their structural integrity.

This analysis will not determine the hydrostatic loading placed on walls due to a flood nor assess when and how these walls would fail. Because the walls at nuclear power plants are typically reinforced concrete and one to three feet thick, it is felt that before a wall would collapse from the flood pressure exerted on it, the component(s) of interest would be submerged. Structural integrity also includes penetration seals and watertight doors. In the cases of doors, the door frame is assumed to remain intact. Doors are treated explicitly in this analysis in that if the flood source seats a non-watertight door against its frame, the door is assumed to stop all but 10% of the flow out of the flood source in the room. Thus, a water level may be built up in the room that can potentially submerge a component. Conversely, if the flood source pushes against a door such that the latch is the only thing that is keeping the door shut, then it is assumed the door immediately opens and passes flood waters.

Note: Watertight doors and other doors are assumed to have a low probability of being open since these doors tend to serve as fire barriers and as such are designed to be self shutting and or are alarmed indicating their status.

e. Flooding via drains backing up into other areas will not be considered.

A detailed analysis of plant drain systems to ascertain and evaluate the chance of drains backing up and causing problems will not be performed. (This would entail knowing flow paths, check valves installed, reliability of these valves, all possible sources that could feed combinations of these drain paths, etc.) However, one aspect of drains will be addressed. The first is that, in all probability, when a flood occurs where there is a drain, some debris will eventually block all or a

0514

portion of the drain. When and how much blockage would actually occur is unknown, but for this analysis it will be assumed that at the inception of a flood, drains are blocked by a factor of 20% of their flow rate (i.e., the drain can handle only 80% of its design capacity). It is also assumed that the location of the ultimate drain source (e.g., a tank or reservoir) is capable of handling the influx of water and does not become a secondary flood source.

f. Only single flood sources will be analyzed.

Although it is recognized that a seismically-induced internal flood could cause multiple pipe breaks to occur in the plant, evaluation of this situation is beyond the scope of TAP A-45. This analysis is limited to single flood sources since they can be examined in a timely and efficient manner and are reproducible and tractable.

g. Components will fail from submergence in water to a critical point or from direct spray from the flood source.

Electrical equipment (primarily motors/pumps, switchgear, and buses) is by and large susceptible to a flood since it causes electrical shorting of the component. Humidity will not be considered as a failure mechanism due to the lack of failure data available and because equipment is assumed to be qualified to the worst environment it will see. (Note: The purpose of TAP A-45 is not to experimentally develop this type of data.) Most purely mechanical components (i.e. those not using electricity as a motive force) are not directly vulnerable to water unless a) the water can cause mechanical damage, such as entering the casing of a turbine and causing blade damage or b) the control systems for these components are electrical and their failure causes the component to malfunction. Mechanical component damage is considered when steam-driven pumps are important front line or support system components.

It is assumed that switchgear and buses will fail if they are standing in six (6) inches of water or more (typically contacts in large breakers are found at this level). In a similar fashion, an electric motor will be assumed to fail when the water level reaches the bottom of the motor casing, unless it can be determined that the motor is designed to be watertight. Cables will be assumed to maintain their integrity if submerged. Consideration of motor operated valves (MOVs) is beyond the scope of TAP A-45 because of the large number of MOVs and the possibility of complex interactions and timing in valve alignments. These assumptions are consistent with those made in previous internal flood PRAs.

h. Random failures of those systems not directly affected by the internal flood at a particular location will be considered if applicable.

For those cases where particular systems are not affected by the flood, random failures of those systems will be evaluated. These would be the systems or trains that for the short term would be expected to provide decay heat removal. This approach ensures that credit is given for these systems but that the potential for their incorrect operation is factored into the analysis (i.e., after an internal flood, a system could still randomly fail).

i. Offsite power is assumed to be available during an internal flood unless the flood is capable of causing a loss of offsite power.

In most situations, it was assumed that given an internal flood and a reactor scram, offsite power would be available to the plant for recovery actions. This assumption only affects those sequences that include the power conversion system in one of the success paths on the event tree. Thus, all possible success paths that are indicated on the transient event trees are included in the analysis.

j. No explicit core melt timing is evaluated.

The TAP A-45 systems analysis includes thermal-hydraulic phenomenology in determining the timing for internal event sequences to determine potential recovery time windows. Typical time windows before core melt ensues are several hours. However, for the internal flood analysis, it is assumed that the repairs that would be required to be made after a flood had occurred would not be able to be accomplished in these time windows because of the multitude of damage that might be expected. Therefore, the frequencies that are determined from this analysis do not reflect heroic actions that might be taken by plant personnel to return some decay heat removal systems to operation prior to the onset of core melt. This assumption is a conservative approach.

Note: The approach used in this analysis is designed to look primarily at the effects of an internal flood on a single unit. The results from the analysis will be presented for one plant, but if applicable, insights on how the internal flood could affect a two unit plant will also be included.

2. APPLICATION OF METHODOLOGY

The implementation of the methodology previously presented will be applied to the St. Lucie, Unit 1 Nuclear Power Plant in this section. This section follows closely the steps previously presented.

0516

2.1 Initial Screening to Determine Potentially Important Fire Areas for further Analysis

Identification of Systems

This step involved the identification of those systems and their associated success criteria from transient event trees developed for the plant that lead to 'OK' states on the event trees. The transient event tree examined was T3, Transient with Power Conversion System Initially Available. This event tree best matches the assumptions used in this analysis as well as the fact that the other two transient event trees constructed for St. Lucie are just modified subsets of this event tree, and T3, as such, contains all the success paths being modeled. From this transient event tree, three potential success paths were identified. These success paths were made up of the following systems or combinations of systems as follows;

a. Power Conversion System (PCS) alone

b. Auxiliary Feedwater System (AFWS)

c. Safety Injection (SI) System in Feed and Bleed mode leading to High Pressure Recirculation (Safety Injection Pumps and Residual Heat Removal (RHR) pumps and Heat Exchangers available)

From this event tree, if any of these systems could function in a transient as designed, the core would not be damaged. Therefore, for this analysis, the three front line systems of importance became combinations of a, b, and c above (see systems analysis section for presentation of event trees). The next step, once the front line systems had been identified, was to determine their success criteria. The success criteria for these systems was obtained from the systems analysts and is summarized below.

a. Power Conversion System: 1 out of 2 trains

b. Auxiliary Feedwater System: 1 out of 2 motor driven pumps or the turbine driven pump train.

c. Safety Injection System: 1 out of 3 trains

d. Residual Heat Removal System (in recirculation mode): 1 out of 2 RHR pumps in series with 1 out of 2 RHR heat exchangers.

For the front line systems the success criteria has an important implication for this analysis. Because in general the success criteria only require that 1 train or pump from each system is needed, to invalidate the success criteria implies that locations must be found in the plant were all trains of a

0517

particular system can be damaged by a fire. More importantly, this means in general that for this analysis, only those locations where all the trains of the systems listed above can be damaged are important. (Note: in general this is true, however locations may exist where a front line system may not have all its trains together, but its support systems may, therefore support systems must be carefully examined also.)

As per the methodology, not only front line systems were examined, but also the support systems required for the front line systems. The systems analysis section identifies the support systems necessary for continued operation of front line systems. In general the systems that are required are:

a. Intake Cooling Water (ICW): Provides cooling to the following: component cooling water heat exchangers, turbine generator cooling systems.

b. Component Cooling Water (CCW): Provides cooling to the following: safety injection pumps, RHR pumps and heat exchangers, and other systems not considered as part of this analysis

c. AC/DC Power: power is provided to all the pumps that have been previously mentioned as well as actuation signals and controls for motor operated values

The success criteria for the support fluid systems are:

a. Intake Cooling Water: 1 out of 3 ICW pumps

b. Component Cooling Water: 1 of 3 CCW pumps with 1 out of 3 CCW heat exchangers

Once again, it can be seen that to invalidate a support system criteria, only those locations that have all of the trains in a system can effectively do this. This information is used in the next step to determine which plant areas need to have a more detailed analysis performed for them.

Screening to determine potentially important flood areas

The next step in the analysis was to determine which areas in the plant could meet the requirement of having enough of either front line or support systems to invalidate all the success paths possible for the plant. This was done by a review of readily available plant documentation and mapping the locations and components (refers to both components and cables) into a matrix, as a first screen, to determine if an area had enough of the combinations of systems to get one into trouble. For this analysis, the utilities latest Appendix R submittal [9] was used to determine the components in a given area. Fire area information was used since in general it corresponds to individual rooms and thus areas where a flood could expect to be contained to some extent.

Table E2 is a matrix of the areas of interest and as a first screen determines if any of the front line or support systems identified by using the above documentation appear in the zone of interest.(Note: this table shows if 1 train is running through a location or if it has all the systems and trains.)

From the table, several zones can be automatically eliminated since they do not contain any of the success oriented systems of interest. These are zones 9, 10, 11, 12, 18, 27, 29, 37, 39, 40, 41, 42, 43, 44A, 45, 46, 48, 49, 50, 51, 52, 53, 62, 64, 65, 66, 67, 68, 69, 71, 72, 73, 74, 79, and 81. Of the zones left, a sizable number of these zones can also be eliminated by looking at the physical layout of the plant.

St. Lucie's physical layout can best be described as a detached system of structures. Specifically, the center of the plant is the reactor containment building. An auxiliary building, called the reactor auxiliary building, next to the containment, contains the RHR, SI, and charging pumps as well as associated heat exchangers. The reactor auxiliary building has, in essence, a large basement area where all these pumps are located. There are curbs installed at the entrance to each pump room, which would impede the flow of water. All pumps are mounted on concrete pedestals at least a foot high so that a considerable volume of water would be required to damage them due to submergence. Because of the large area involved, a "large" volume of water would be required to cause any damage to these pumps.

A control structure, housing the control room, cable spreading room, battery rooms and other electrical equipment is located on top of the reactor auxiliary building structure. These areas are above most tanks found at St. Lucie. The only major sources of water found in these areas are for fire fighting purposes (i.e., manual hose systems, no automatic water suppression systems are used at St. Lucie Unit 1).

The main power producing turbine generators are located in a separate structure. The "main turbine deck" contains the main turbine and its associated systems (i.e. condensate, feed, etc.). This structure has multiple openings to the outside so that any internal flood would not be able to build up any appreciable water level. Also found in this general area are the auxiliary feedwater pumps, which are essentially out in the open.

The physical separation between the various safety components of interest for this analysis is on the order of several hundred feet. Specifically, the AFW pumps, and SI and RHR pumps are widely separated, and because the AFW pumps are housed in structures that are open to the outside, any

0519

potential internal flood would not have the capability to physically damage both of these systems. (In a similar fashion, the same can be said for the PCS and AFW systems). (Note: a flood in the auxiliary building, if located near or in the RHR pump rooms, could damage these pumps since the RHR pump rooms are at the lowest level in the plant. However the AFWS and PCS success paths would not be affected.)

Support systems for front line systems (i.e. CCW and ICW) are all located in outside areas. The CCW pumps are located behind the reactor containment building and are mounted on pedestals several feet high. The area is completely exposed to the environment, thus internal flooding could not affect these pumps. Similarly, the ICW pumps are located several hundred feet from the reactor auxiliary building and are also exposed to the outside environment. Both the CCW and ICW are specifically designed for exposure to rain and are housed in NEMA enclosures.

Because of the wide separation found between the various systems, the only areas that could reasonably be expected to have the capacity to cause all success paths to be invalidated would be electrical distribution or control centers found in enclosed structures. Thus, the number of potential zones to be examined in more detail are the following:

Zone	Description
36	Main Hallway (- 0.5' elev.)
44A	'A' Cable Loft Enclosure
55E	Main Hallway (elev. 19.5') East
55W	Main Hallway (elev. 19.5') West
56	Class 1E 'B' Switchgear Room
57	Battery Charger, Inverter, and Cable Spreading Room
60	Class 1E 'A' Switchgear Room
70	Control Room

(Either individually or in combination with other zones, the zones listed above could lead to a core damaging situation.)

2.2 Quantification of Selected Zones

This section will quantify the risk due to an internal flood in the zones of interest that can lead to core melt. Note that in this quantification certain assumptions are used (as previously presented) that may terminate the analysis steps. If it is shown, based on the assumptions being used, that no damage would occur to the components in a particular zone(s) that would invalidate the success criteria, then the analysis will not proceed and it will be assumed for this analysis that the residual risk for such an area is less than 1.0E-06/Rx yr of leading to potential core melt and as such the cost/benefits would be minimal for any proposed modifications in this zone.

2.2.1 Tank Induced Flooding Sources

One aspect of an internal flood that must be addressed is if tank failures could lead to the flooding of enough of the zones listed to invalidate the success criteria. Therefore, the first cut taken at determining if a flood could cause damage was to postulate tank ruptures and using their nominal fluid volumes determine a. if the volume of flood water would be capable of flooding out enough of the zones and b. the flow paths that the fluid in all likelihood would take. The volume and path taken by the fluid upon rupture was determined by plant drawings and FSAR data. As an initial screen, it was assumed that each tank examined could completely empty its contents into a room and that the room would fill up to an appropriate level based on the tank volume. This assumed that each room was watertight and was not open to any other areas. Using the 20% derating factor for the floodable volume for each of the rooms identified above, the following tanks were assessed as being able to flood at least one of the zones previously identified, sufficiently high (.305 meters), to cause damage if ALL THE TANK CONTENTS WERE EMPTIED INTO ONLY THE ONE AREA.

a. Reactor water storage tank (RWST)
b. Condensate storage tank
c. Main condenser (not actually a tank; but a large reservoir)
d. Primary water storage tank
e. Hold up tanks
f. Fuel pool
g. Aerated waste storage tank
h. Caustic storage tank

All other tanks were found to have insufficient volume to be able to fill any of the areas up to critical damage heights. Next, the flow paths that the water could take from the tanks to affect the areas previously identified was determined. This was done by using the plants general arrangement drawings and looking for openings, stairwells, and gratings by which the water could spread from room to room. It should be noted that at this step none of the components of interest were under open gratings or openings such that dripping water could lead to damage. Therefore all damage had to occur due to rising water levels. As mentioned previously, most of the critical locations were found above ground level in the plant (except the Main Hallways, where cables were of concern versus components). In addition, because all the rooms of interest were electrical distribution centers, the critical water level used in the analysis was .305 meters (i.e., if this level was reached, failure was assumed to have occurred). The results of this tank analysis showed that no rooms or areas could be flooded out by the tanks previously mentioned. The RWST, condensate storage tank, and primary water storage tanks are all physically located outside, therefore their rupture would not be able to affect the areas of interest inside the plant. These tanks are located at ground level, therefore any tank induced flood waters would spread out over a large area. There would only be a thin layer of water which would not meet the .305 meter flood level damage limit. (This plant has at least .305 meter high curbs at entrances into inside plant areas, therefore flood waters would need to be at least this high to cause any problems.)

The fuel pool would not cause flooding problems in the critical flood locations since it is physically separated from the reactor auxiliary building where all the critical flood locations are found. The rupture of the main condenser would cause the loss of the PCS system , however, the AFW pumps would still be available since they are located some distance from the main condenser in essentially an open ground level area, allowing water to disperse before it could damage the AFW pumps. In addition, the SI and RHR systems would still be available.

The Aerated Water Storage Tank, located below ground level (elev. -.5') could only affect the Main Hallway (elev. -0.5'). However, the volume of water found in this tank as well as the large surface area of the Main Hallway would prevent the critical .305 meter high water depth required to cause damage from being reached.

A similar rationale applies to the Caustic Storage Tank. It causes less of a problem, however, than the Aerated Waste Storage Tank since it has a smaller volume. The only other tank that could potentially cause a problem in any of the critical flood locations is a Hold Up Tank. These tanks, which are approximately 50 feet high, are located in individual plant enclosures. There are two access doors to these tanks--one at elevation 43.5' and one at elevation 20'. The access at 43.5' would not pose a problem to the plant since this access point is above the maximum height of water that could be developed in the enclosure if the tank ruptured. At the 20' elevation, a tank rupture could potentially spill out into the Main Hallway (elevation 19.5') and potentially flow down to the Main Hallway (elev. -0.5'). (All the other critical flood locations are physically located above the 19.5' elevation.) However, in neither case would the depth of water be greater than the .305 meter critical water damage height, therefore a Hold Up Tank would not lead to failure of any of the critical flood locations.

Neither the Auxiliary feedwater nor the SI/RHR systems would be affected by these tank induced floods, therefore internal flooding due to tank ruptures at St. Lucie appears to not be a major concern.

0522

2.2.2 Flood Sources found outside the zones of interest

In this portion of the internal flood analysis,the largest flood sources found outside the areas of interest were determined and evaluated as to if they could lead to enough damage to invalidate all the success paths. The principal sources found outside of these areas were:

a. Circulating water for the main condensers
b. Intake Cooling Water
c. Fire main piping

As in the tank case, a rupture of the piping was postulated for this piping on each level of the plant and then flow paths were determined to see if enough of the systems could be damaged. Using the assumptions outlined above, it was found that for a postulated circulating water break at the main condensers (being the largest flood source on the ground level of the plant) that although the PCS system would be lost, for the same reasons as given in the tank analysis, the other critical areas would not be affected. In addition, it was felt that a flood of this nature would cause plant personnel to turn off the circulating water pumps rapidly and therefore there would not be an endless supply of water being pumped into the turbine area but rather a short gush of water that would then spread out over a large area. (Even if this did not take place, because the turbine structure is detached from the rest of the structures at St. Lucie, water would run off into the surrounding canals and streams without any appreciable depth of water being formed in any of the critical areas).

The other locations in the plant only had ICW and the fire main as principal flood sources on the various plant levels. It was assumed, based on discussions with plant personnel, that a ICW main rupture would cause a reactor trip, but that it would be isolated and because of the fact that the ICW header at St. Lucie is a loop system, with multiple cross connects, that cooling to the systems previously mentioned could be restored rapidly. Unless the ICW induced internal flood caused damage initially to components or their switchgear, then they would be available. Because the PCS is not dependent on ICW, it would be available. In addition, ICW runs primarily on the ground level of the plant, thus the cable spreading room, control room, and switchgear rooms would not be affected. Only the Main Hallways would be affected. At ground level water would disperse to open areas via the doors to the outside in these hallways, thus preventing any appreciable water buildup.

In a similar fashion, the fire mains found at the plant were all smaller than the ICW headers running through the plant and therefore since the ICW headers could not cause sufficient damage, neither could the fire main headers. Note: this refers to fire mains found in areas OTHER than the areas that have been identified as critical areas.

2.2.3 Largest flood source in each of the areas of interest and potential to affect that zone and other zones.

All the areas of concern have internal flood sources that are either smaller size flood sources than for those presented above, or they do not have any flood sources at all in the area. In addition, where a flood source exists in an area it either could not cause damage or has been provided with special protective measures making the possibility of an internal flood in an area remote or redundant capabilities are available elsewhere. The 1A and 1B switchgear rooms fall into this latter case in that they have small water sources which might damage equipment by water spray, but because they are redundant to each other, the loss of one switchgear room still would provide the plant with at least one half of all safety system trains available.

The control room does have small flood sources due to kitchen and lavatory areas that are recessed off the main control room area, however, the sink area is separated by two sets of doors and is not in a direct line of sight with cabinetry so that the question of spray damage is not applicable. The chance of getting six inches of water into the cabinets is deemed incredible from this source. The cable spreading room also has a small flood source from a chilled water cooler, however, the only important safety components in this room are cables and DC distribution panels and chargers, so spray from this source would have limited damaging effects (i.e. cables are not damaged by flood sources and since one ESF DC distribution system is enclosed in its own room in this area, one train of ESF DC would be available). Thus from this analysis, for the electrical distribution and control centers of interest, flood sources within the actual rooms are not capable of causing or leading to a core-melt situation.

3. CONCLUSIONS AND INSIGHTS

At St. Lucie it appears that none of the plant areas are susceptible to internal flooding and/or spray. This is primarily due to the fact that the physical layout of the plant is such that almost all potential flood sources are found at ground level, while most of the critical flood locations are found above ground level. Thus there is in essence zero interaction between these critical locations and internal flood sources.

Insights: Specific plant and uncertainty insights are presented below:

Plant Insights:

a. The analysis for Unit 1 is applicable to Unit 2.

0524

b. Because of the design of the plant, internal floods originating in the SI pump area cannot affect the AFW pump area since it is in a different location (i.e. separated by several hundred feet). However, if an internal flood could affect the AFW pump area, it could also potentially affect the 4160 Volt Switchgear room which is in close proximity to the AFW pump area and in turn could cause the loss of SI pumps for both units. The most likely flood source that could cause this scenario would be the main condenser. However, because of the openness of the plant design, the probability that the scenario described above would have serious consequences on all decay heat removal systems is probably insignificant.

c. The only areas in the plant that could be susceptible to flooding are the RHR and SI pump rooms. These rooms are at the lowest elevation in the plant and any flood waters that reached these rooms would stay there. However, the chance of these rooms being affected is small since the pumps are on elevated pedestals, there is a one (1) foot high curb protecting entry into the room and potential flood sources (such as a holding tank or fire main) would first have to flood the hallway area in the reactor auxiliary building at elev. -0.5' to a depth of 1'. To then flood the pump rooms, which have a large surface area, to approximately the same depth, would cause a tremendous volume of water, not readily available from internal flood sources.

Uncertainty Insights

Some of the potential major areas for uncertainty are as follows:

a) The potential for spray sources and/or submergence to actually cause damage to equipment.

b) The effects of humidity from an internal flood on safety-related equipment.

c) The method of partitioning used to determine generic room or building internal flood frequency from the internal flood occurrence data base. Similarly, the applicability of generic internal flood frequency data to an individual plant. Also included in the uncertainty of internal flood frequencies is the uncertainty associated with the size of the source of the internal flood.

d) The human factors uncertainty associated with whether the operators will correctly isolate the flood source.

e) The uncertainty associated with the performance of doors, dampers, and penetrations during an internal flood.

4. REFERENCES

1. "Seabrook Station Probabilistic Safety Assessment", Section 9.5, Public Service Company of New Hampshire and Yankee Atomic Electric Company, December 1985

2. Kazarian, M. and K. Fleming, "Internal and External Flood Failure Model", Chapter 10 of The Impact of External Events, Fire, Internal Flood on Nuclear Power Facilities, Notes from ACTA Consulting Engineers and Applied Scientists, 1984.

3. "St. Lucie Plant Unit 1 Plant- Unit No. 1, Fire Hazard Analysis, Florida Power and Light Company, April 1983 (supplemented with "Nuclear Fire Protection Program, St. Lucie Unit 1 Plant, Unit No. 2," Florida Power and Light Company, January 1985).

4. Alternate Shutdown Report for St. Lucie Unit 1, Florida Power and Light Company, October 7, 1983, letter to NRC.

5. TABLES

Table E1

Internal Flood Frequency as a Function of Location and Severity [2]

Frequency Location	Severity Level	Mean Flooding per Reactor Year
Turbine Building	Moderate	1.0E-2
Turbine Building	Large	8.1E-3
Turbine Building	Very Large	7.7E-3
Auxiliary Building	Small	4.4E-3
Auxiliary Building	Moderate	2.2E-2
Auxiliary Building	Large	1.2E-2

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Table E2

Zone Matrix

Fire Zone	PCS	AFWS	SI	RHR	ICW	CCW	PORV	AC/DC
1 (Steam Trestle Area)	X	X						
2 (Condensate Storage Tank Enclosure)		X						
3 (Intake Cooling Water Area)					X			
4 (Diesel Oil Storage Tank Area)								X
5 (Component Cooling Area)					X	X		X
6 (Diesel Generator 1A Room)								X
7 (Diesel Generator 1B Room)								X
8 (Refueling Water Tank)			X		X			
9 (Primary Water Storage Tank Area)								
10 (Gas Storage Building)								
11 (Miscellaneous Oil Storage Building)								
12 (Water Plant)								
13 (Turbine Lube Oil Reservoir Area)	X							
14 (Condensate Pump and Condenser Area)	X							
15 (Condensate Pump Pit)	X							
16 (Main, Auxiliary and Startup Transformer Area)	X	X			X			X
17 (Turbine Cooling Water Pumps and Heat Exchanger Area)	X							
18 (Instrument and Station Compressors Area)								
19 (Feedwater Pumps 1A and 1B)	X							
20 (Heater Drain Pump Area)	X							
21 (Non-Class 1E 'A' Switchgear Room)	X							
22 (Non-Class 1E 'B' Switchgear Room)	X							
23 (L.P. Heater Area)	X							

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0528

Table E2 (Continued)

Zone Matrix

Fire Zone	PCS	AFWS	SI	RHR	ICW	CCW	PORV	AC/DC
24 (Isolated Phase Bus Area)	X							
25 (Turbine Deck)	X							
26 (Containment)			X				X	
27 (Aerated Waste Storage Tank Area)								
29 (Chemical Drain Tank Area)								
31 (Shutdown Heat Exchanger 1B Room)				X				
32 (Shutdown Heat Exchanger 1A Room)				X				
33 (Piping Penetration Room)				X				
34 (1A Emergency Core Cooling System Room)			X	X				
35 (1B Emergency Core Cooling System Room)			X	X				
36 (Main Hallway (-0.5' elev))		X	X	X	X	X		X
36A (Charging Pump Access Hallway)			X					
37 (Boric Acid Condensate Pump)								
38 (Charging Pump 1C Cubicle)			X					
39 (Gas Decay Tank Area)								
40 (Boric Acid Make-up Tank Room)								
41 (Hold-up Tank Enclosure)								
42 (Chemistry Personnel Building)								
43 (Control Building Personnel Area)								
44 (Radio Chemistry Lab)				X				
44A ('A' Cable Loft Enclosure)		X	X	X	X	X		X
45 (Piping Penetration Room)								
46 (Containment Purge Room)								
47 (Class 1E 'AB' Switchgear Room)			X		X	X		X
48 (Letdown Heater Exchange Room)								

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0529

Table E2 (Continued)

Zone Matrix

Fire Zone	PCS	AFWS	SI	RHR	ICW	CCW	PORV	AC/DC
49 (Volume Control Tank Room)								
50 (Demineralizer Area)								
51 (Drumming Station)								
52 (Boric Acid Water Concentrator Cubicles)								
53 (Laundry and Decontamination Area)								
55E (Main Hallway (elev. 19.5) East)	X	X	X	X	X	X		X
55W (Main Hallway (elev. 19.5 West)	X	X	X	X	X	X	X	X
56 (Class 1E 'B' Switchgear Room)	X	X	X	X	X	X	X	X
57 (Battery Charger Inverter & Cable Spreading Room)	X	X	X	X	X	X	X	X
58 (Battery 1B Room)								X
59 (Battery 1A Room)								X
60 (Class 1E 'A' Switchgear Room)	X	X	X	X	X	X	X	X
61 (HVAC Equipment Room)					X			
62 (Resin Addition Tank Area)					X			
63 (Closed Blowdown Heat Exchanger Area)	X							
64 (Fuel Pool Pumps and Heat Exchanger Area)								
65 (New Fuel Container Storage Area)								
66 (Spent Fuel Pool Area)								
67 (Cask Wash down Area)								
68 (New Fuel Storage Area)								
69 (Spent Fuel Pool HVAC Area)								
70 (Control Room)	X	X	X	X	X	X	X	X
71 (Control Room HVAC Area)								
72 (Auxiliary Building Roof)								
73 (Technical Support Center and Offices)								
74 (Spent Resin Tank Room)								
75 (Charging Pump 1B Cubicles)								
76 (Charging Pump 1A Cubicle)			X					

530

Table E2 (Continued)

Zone Matrix

Fire Zone	PCS	AFWS	SI	RHR	ICW	CCW	PORV	AC/DC
77 (East Electrical Penetration ('A' Train Room)			X			X		X
78 (West Electrical Penetration ('B' Train Room)	X		X				X	X
79 (Raw Water Storage Tank Fire Pump Area)								
80 (Steam Generator Blowdown Tank Room)	X							
81 (Area Between Auxiliary Building and Containment)								
82 (Component Cooling Water Surge Tank Room)						X		

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APPENDIX F

EXTERNAL FLOOD ANALYSIS FOR THE ST. LUCIE NUCLEAR POWER PLANT

by

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0532

TABLE OF CONTENTS

Page

1.0 INTRODUCTION F-1 534
    1.1 Background F-1 534
    1.2 External Flood Risk Analysis F-1 534
    1.3 Analysis Approach F-3 536
    1.4 Report Organization F-5 538

2.0 PLANT AND SITE CHARACTERISTICS F-6 539
    2.1 Plant Location and Flood Hazards F-6 539
    2.2 Plant Description and Flood Design Basis .. F-8 541
    2.3 Vulnerable Structures and Equipment F-9 542

3.0 FLOOD HAZARD ANALYSIS F-19 552
    3.1 Screening Analysis F-19 552
    3.2 Hazard Assessment F-24 557
        3.2.1 Methodology F-24 557
        3.2.2 Results F-28 561

4.0 FRAGILITY ANALYSIS F-37 570
    4.1 Methodology F-37 570
    4.2 Results F-40 573

5.0 SYSTEM ANALYSIS F-45 578
    5.1 Methodology F-45 578
    5.2 Results F-46 579
        5.2.1 Component/System Failures Due to
              External Flooding F-46 579
        5.2.2 External Flooding Event Tree F-49 582

6.0 RISK QUANTIFICATION F-51 584
    6.1 Methodology F-51 584
    6.2 Results F-52 585

7.0 REFERENCES F-53 586

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1.0 INTRODUCTION

1.1 Background

Sandia National Laboratories (SNL) is participating in the Task Action Plan (TAP) A-45 program to evaluate the adequacy of current licensing requirements for shutdown decay heat removal systems. As part of the SNL study, the response of decay heat removal systems to internal and external initiating events is being considered. The full scope of the analysis plan is reported in Reference 1.

Science Applications International Corporation (SAIC) is assisting SNL in the internal event and external event analyses being conducted for the U.S. Nuclear Regulatory Commission (USNRC). Jack R. Benjamin and Associates, Inc. (JBA) is a subcontractor to SAIC and provided the flood hazard and fragility data used in the TAP A-45 risk analyses. The external flood analyses for the St. Lucie Unit 1 (referred to as St. Lucie) are given in this report.

1.2 External Flood Risk Analysis

As stated in Reference 1, it is beyond the scope of the TAP A-45 program to perform an in-depth probabilistic risk assessment. Instead, the objective is to conduct an analysis that quantifies the significant threats to a plant. The analysis procedure used to perform the flood analysis embodies the basic philosophy of full-scope probabilistic risk assessment.

To evaluate the frequency of plant damage due to external flood events, the following tasks are performed: 1) plant familiarization, 2) hazard analysis, 3) fragility/vulnerability evaluation, 4) systems analysis, and 5) risk quantification. Figure 1-1 shows the relationship of these elements. The systems analysis and risk quantification were performed by SAIC and SNL, while the hazard and fragility analyses were conducted by JBA.

0534

The steps in the analysis are summarized below.

- Plant Familiarization - The purpose of this task is to gather information on the occurrence of external flood hazards at a plant and on the vulnerability of plant structures and equipment to external flooding. In assessing the reliability of shutdown decay heat removal systems, a review of available documentation (i.e., FSAR, plant drawings and topical reports) and a walkdown of the plant are performed.

- Hazard Analysis - In the case of external flooding, the hazard analysis is conducted in two steps. First, a screening of the potential sources of flooding (i.e., upstream dams, local precipitation, storm surge, etc.) in the vicinity of the plant is made. For sources of flooding that can impact the plant site, an analysis is conducted to evaluate the frequency of occurrence of increasing levels of flood intensity (i.e., depth of inundation, hydrodynamic forces, etc.). An uncertainty analysis is also performed to assess the modeling or professional uncertainty in estimating the frequency of occurrence.

- Fragility Evaluation - For structures and equipment items vulnerable to flood hazards, the fraction of failure defined as a function of a flood hazard parameter (i.e., depth of inundation), called a fragility curve, is assessed. The modeling uncertainty in the evaluation is also considered.

- Systems Analysis - The shutdown decay heat removal systems analysis is performed by developing event trees and fault trees with external flooding as the initiating event. The component fragilities are used to evaluate the conditional fraction of failure of different safety systems.

- Risk Quantification - The quantification of the likelihood of plant damage is assessed by properly weighting the conditional fraction of plant damage (i.e., system fragility curve) defined as a function of flood intensity, by the frequency of occurrence of each flood level. By propagating the uncertainty in the flood hazard frequency assessment and the component fragilities, the probability distribution on the frequency of core melt is evaluated.

The scope of this report includes all of the above tasks.

1.3 Analysis Approach

To evaluate the contribution to core melt of external flood hazards, a simplified flood risk assessment approach was used. The flood risk analysis performed as part of TAP A-45 relied heavily on available information on regional and local flooding and any studies performed specifically for the plant. A screening procedure was used to eliminate sources of flooding as potential risk contributors where conservative arguments indicated this was reasonable. Within the scope of the TAP A-45 program, simplified procedures were then used to evaluate the frequency of occurrence of flood hazards that could impact the plant.

A conservative approach was used in developing the capacities of the structures and equipment to resist the effects of flooding. This procedure was based on the a prior/ assumption that the individual contribution of each of the components to core melt was not significant. When it was found in the systems analysis that external flooding was a significant contributor relative to other external and internal event initiators, then a more detailed look at components which dominate the flood contribution was required. By proceeding in this manner the analysis effort focused on the most significant contributors, and components which were not contributors were screened out by a defensibly conservative approach that required a minimum of effort.

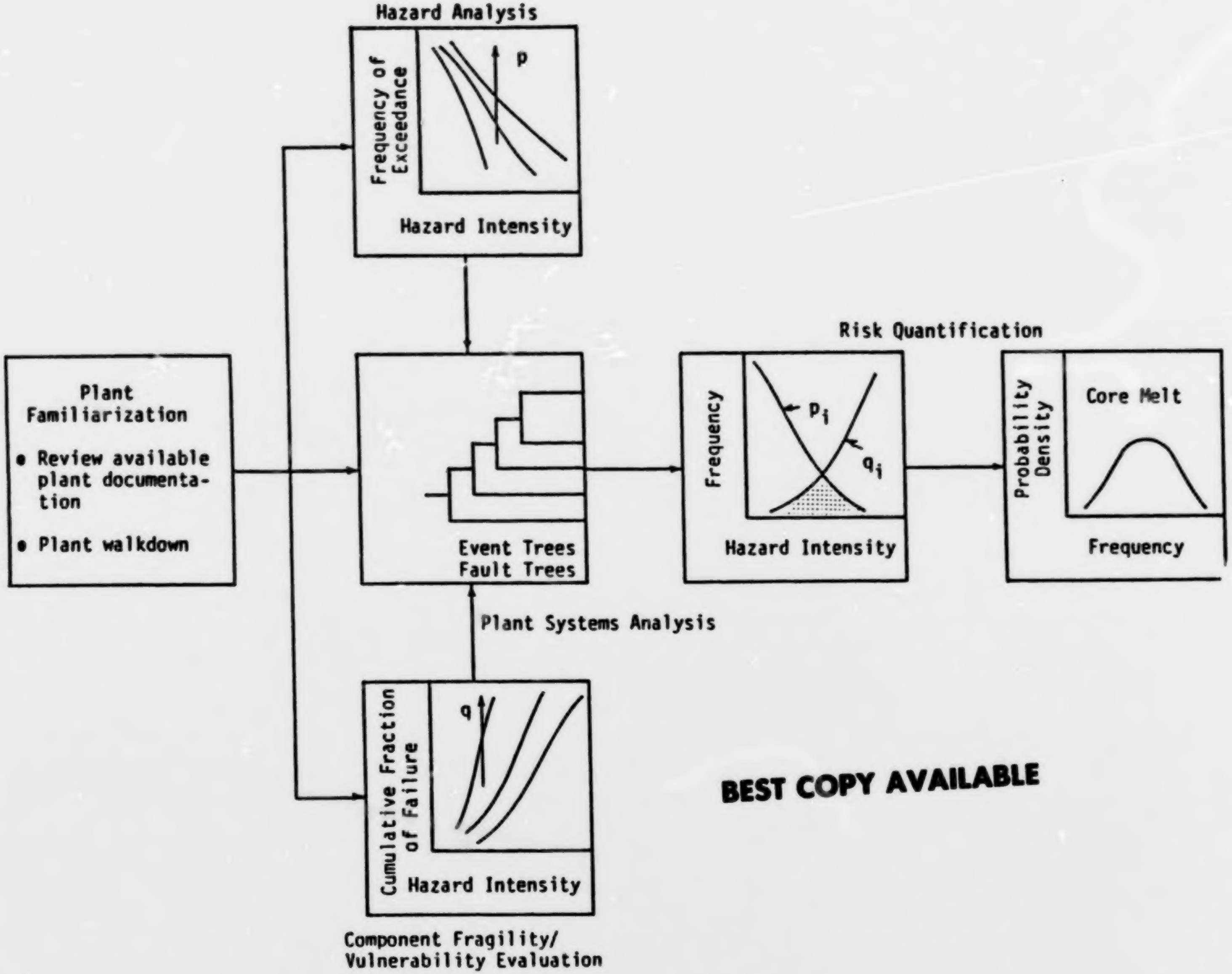
Hazard Analysis
Frequency of Exceedance
p
Hazard Intensity
Plant Familiarization
• Review available plant documentation
• Plant walkdown
Event Trees
Fault Trees
Plant Systems Analysis
Risk Quantification
Frequency
p_i
q_i
Hazard Intensity
Probability Density
Core Melt
Frequency
Cumulative Fraction of Failure
q
Hazard Intensity
Component Fragility/
Vulnerability Evaluation

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FIGURE 1-1 RISK ANALYSIS FLOWCHART

1.4 Report Organization

In Section 2.0 the plant and site characteristics of St. Lucie are described from the perspective of the affects that external flood events have on shutdown decay heat removal systems. The plant site is described and the structure and equipment items vulnerable to external flooding are given.

In Section 3.0 the assessment of external flood hazards is discussed. Based on an initial screening of the possible sources of flooding, an assessment of the frequency of flooding at the plant site is made.

In Section 4.0 the results of a fragility evaluation of structures and equipment exposed to on-site flooding are reported.

In Section 5.0 the modeling of the plant's systems responses is performed.

In Section 6.0 the risk quantification is performed to determine the contribution to core melt frequency due to external flooding.

2.0 PLANT AND SITE CHARACTERISTICS

In this section a description of the St. Lucie plant and site characteristics is given. As part of the flood analysis, a plant walkdown of St. Lucie was conducted to gain familiarity with the plant and site characteristics and to evaluate the vulnerability of structures and equipment important to decay heat removal to external flood hazards. Plant design documents including the Final Safety Analysis Report (FSAR) (Ref. 2) and construction drawings were obtained and reviewed. In addition, discussions with plant and utility personnel were conducted to obtain an understanding of the protection provided to resist the effects of flooding.

In Section 2.1 the plant location is discussed from the perspective of the external flood analysis, including a general description of the flood hazards at St. Lucie. In Section 2.2 a discussion of the plant characteristics with respect to external flooding is given. A summary of the structures and equipment vulnerable to flood hazards is given in Section 2.3.

2.1 Plant Location and Flood Hazards

The St. Lucie plant is located on Hutchinson Island in St. Lucie County, state of Florida. The plant site is about halfway between Ft. Pierce and Stuart, and approximately 80 miles north of Miami Beach. The Atlantic Ocean serves as the eastern boundary for the site whereas the western boundary is the Indian River, a salt water estuary. Figure 2-1 shows the location of the plant and the topography of the plant site area.

In the vicinity of the plant the terrain is characterized as a low lying salt marsh. The local topography consists of a dune along the Atlantic shore which varies in elevation from about 8 to 14 feet above

Mean Low Water (MLW). Between the dunes and the plant there is a salt marsh, approximately 1200 feet wide with an average elevation of 2 feet MLW. The salt marsh is bounded on the west by state highway A1A which passes in front of the plant and has a crown elevation of 18.3 feet MLW. Plant structures are founded on engineered fill constructed to an elevation of 18 feet above MLW. In the immediate vicinity of plant structures, grade is 19 feet MLW.

The principal causes of extreme flooding in the vicinity of St. Lucie can be attributed to storm surge and wave action due to tropical storms and hurricanes. During extreme tides and tropical storms or hurricanes, the low lying areas surrounding the plant site are inundated. In addition, rainfall can cause ponding in the low lying areas. However, high soil permeability provides significant ground water recharge from local precipitation, with little runoff (Refs. 2 and 3). On site, runoff is provided by a storm sewer system and surface runoff to low lying areas and to the intake and discharge canals.

Tidal flooding due to the surge produced by tropical storms has occurred on a number of occasions in the St. Lucie area. Historically, the average frequency of occurrence of tropical storms (including hurricanes) that cross the coast (i.e., landfalling storms only) within 50 miles of St. Lucie is 0.17 events per year (Ref. 4). Of these storms, approximately 70 percent are hurricanes. The frequency of occurrence of exiting storms occurring in proximity to St. Lucie is less than the rate of landfalling storms (Ref. 4).

Table 2-1 lists flood elevations which have been recorded in the St. Lucie area. The maximum tidal surge observed closest to the plant was 8.5 feet Mean Sea Level (MSL) which occurred in 1949. (Note, MLW = MSL + 1.3.)

Warning is available in the event of a hurricane. Hurricanes are tracked by the National Hurricane Center, a branch of the National

Oceanic and Atmospheric Administration (NOAA). St. Lucie has emergency procedures in place to respond to a hurricane warning (Ref. 5).

According to the St. Lucie FSAR (Ref. 2), flooding can occur in the vicinity of the plant site due to rainfall and wave-runup. Principally, this will entail ponding in catch basin drainage areas. It is concluded in the FSAR that local flooding due to rainfall is a relatively minor hazard since the bottom of doors to safety-related buildings are at least 0.50 feet above the high points of the drainage areas.

2.2 Plant Description and Flood Design Basis

Figure 2-2 shows a view of the general arrangement of plant structures and outside equipment items at St. Lucie. Safety-related equipment which is part of the shutdown decay heat removal system is located in the Reactor, Reactor Auxiliary and Diesel Generator buildings, and on the Intake Structure. In addition, a number of components such as the switchyard, the condensate storage tank, and the diesel fuel oil tanks are located outside.

The St. Lucie plant is designed to withstand the effects of a storm surge that results from the Probable Maximum Hurricane (PMH). According to Reference 2, the PMH produces a stillwater storm surge of 16.22 feet MLW and wave heights in the discharge canal greater than 10.0 feet. Subsequent to the design of Unit 1, the FSAR for Unit 2 was published (Ref. 3). According to Reference 3, the design basis stillwater surge elevation due to the PMH is 17.2 feet MLW. While wave heights could reach 13.4 feet in the discharge canal, Reference 3 concludes that wave runup on the plant site generally does not exceed plant grade of 18.5 feet MLW. However at the nose of the discharge canal (i.e., the point where the discharge canals for the two units meet) (see Fig. 2-2), wave run-up was calculated to reach 28 feet MLW assuming the embankments were that high. Instead, the runup simply overtops the embankment which has a top elevation of 18.5 feet MLW. According to the Unit 2 FSAR (Ref. 3), runup at this location does not affect safety-related equipment since

2.3 Vulnerable Structures and Equipment

In the event that extreme flooding should occur due to a hurricane storm surge, safety-related components outside would be inundated if the flooding is at least above elevation 19.5 feet MLW. Components which are outside may be inundated by wave action or wave runup.

In the event of a hurricane, emergency procedures are designed to protect plant structures and equipment by closing exterior doors. For a flood which exceeds the height of doorways, 19.5 feet MLW, safety-related equipment could be flooded, if leakage occurs.

Based on a review of plant documents and a walkdown of the plant, an assessment of the location and vulnerability of safety-related equipment was made. Table 2-2 lists the vulnerabilities for each of the structures and equipment considered. It was assumed in the development of the vulnerable structure and equipment list that concrete walls of Category I structures are adequate to protect against hydrostatic or wave forces (Ref. 2). In the case of the Reactor Building it is assumed that it is leakproof; and therefore, safety-related components located inside are not vulnerable to flooding.

The switchyard is situated west of the discharge canal and the main plant site, at an elevation of approximately 18 feet MLW. In the event of a severe hurricane, which is capable of producing flooding on site, it is anticipated that due to the hurricane-force winds, off-site power will be unavailable.

it is more than 400 feet from the nose of the canal. As a result water which overtops the canal embankment at the nose simply runs off the side slopes returning to the discharge canal. At the nose of the discharge canal a sheetpile bulkhead is provided to protect against wave-runup. The top of the bulkhead reaches elevation 22 feet MLW.

The surge associated with the PMH and accompanying wave action would isolate the plant and result in wave runup above plant grade of 19.0 feet MLW. To mitigate the potential for flooding plant structures and equipment, flood protection measures have been taken.

According to the St. Lucie FSAR (Refs. 2 and 3), flood protection at the plant is provided to an elevation of at least 19.5 feet MLW. With one exception, all equipment items are protected to elevation 19.5 feet MLW. Also, through an arrangement of plant structures and a floodwall, openings to plant structures are protected from the direct effects of extreme storm surge or wave runup. At the time of the plant visit, emergency procedures for flooding called for stop logs to be inserted at flood wall openings (Ref. 5). Thus, complete closure of the flood wall provided protection to elevation 22 feet MLW. However, a subsequent visit by SNL reported that a change in plant emergency procedures indicates that the floodwall is not required, resulting in removal of all stop logs.

At the Intake Structure, safety-related equipment is located above elevation 22 feet MLW, except for the motor operators of isolation valves I-MV-21-2 and I-MV-21-3. Since these components are not required to function during a major flood, their potential vulnerability is not critical (Ref. 2).

The two diesel oil storage tanks are protected up to elevation 24.5 feet MLW. Structures that house safety-related equipment are constructed of reinforced concrete and have been conservatively designed to withstand the effects of hydrostatic and buoyant forces associated with a PMH surge elevation of 16.2 feet MLW (Ref. 2).

Should flooding occur beyond the floodwall and enter plant structures, safety-related equipment located on the ground floor (elevation 19.5 feet MLW) or at lower elevations is not always protected in watertight rooms.

As part of the plant design basis, consideration was given to the possibility of flooding due to intense local precipitation. Apparently, two analyses were performed. The first analysis was conducted as part of the Unit 1 FSAR (Ref. 2). Subsequently, a second analysis was performed as part of the Unit 2 safety analysis (Ref. 3). The principal difference in the two studies was the estimate of the Probable Maximum Precipitation (PMP). In Reference 2 the 24-hour PMP is estimated to be 32 inches while in Reference 3 the PMP is 47.1 inches. In both studies a peak rainfall intensity of 6 inches per hour was considered.

The roofs of structures housing safety-related equipment have been designed to handle rainfall runoff. Measures incorporated in the design include limited parapet wall heights, sloping roofs and drainage by exterior leaders to the storm water drainage system. The site storm drainage system and roof leaders are designed for a rainfall intensity of 6-inches per hour. For more intense rainfall, water could collect and ultimately overtop the roof. The accumulation of water on building roofs does not exceed 2 inches, with the exception of the Reactor Building which has a 1.5 foot parapet wall.

According to the FSAR, ponding which could occur on site does not impact safety-related equipment since the top elevation of any catch basin is anywhere from 6 to 18 inches below the lowest door elevation (19.5 feet MLW). As a result any additional inflow to these areas will overtop the low point in the basin and runoff the nuclear island. The FSAR concludes that the extent of local ponding on site is self limiting and does not impact safety-related equipment or structures.

TABLE 2-1

HISTORIC FLOOD ELEVATIONS RECORDED NEAR ST. LUCIE

(From Ref. 2)

Date	Elevation* (feet-MSL)	Record Location
September 1928	10.1	Palm Beach
September 1947	6.8	Indian River, in the vicinity of Ft. Pierce
August 1949	8.5	Stuart, near St. Lucie Inlet

* MLW = MSL + 1.30 feet

0545

TABLE 2-2

VULNERABLE STRUCTURES AND EQUIPMENT

Structure/Equipment	Location	Comment
Containment Spray Pumps	Located in the Reactor Auxiliary Building at elevation -0.5 feet MLW.	Surrounded by interior walls with top elevation +7.50 feet MLW. Doors to the room are watertight and kept closed at all times. Room is watertight, therefore components are not vulnerable to external flooding.
High Pressure Safety Injection (HPSI) Pumps	Located in the Reactor Auxiliary Building at elevation -0.5 feet MLW.	Surrounded by interior walls with top elevation +7.50 feet MLW. Doors to the room are watertight and are kept closed at all times. Room is watertight, therefore components are not vulnerable to external flooding.
Low Pressure Safety Injection (LPSI) Pumps	Located in the Reactor Auxiliary Building, at elevation -0.5 feet MLW.	Surrounded by interior walls with top elevation +7.5 feet MLW. Doors to room are watertight and kept closed at all times. Room is watertight, therefore components are not vulnerable to external flooding.
Shutdown Heat Exchanger	Northwest side of Reactor Auxiliary Building, at elevation -0.5 feet MLW.	Surrounded by interior walls with top elevation +6.5 feet MLW. Doorways are not watertight. Water can run down the basement if flood level is greater than 19.5 feet MLW.

0546

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TABLE 2-2 (Continued)

VULNERABLE STRUCTURES AND EQUIPMENT

Structure/Equipment	Location	Comment
Auxiliary Feedwater Pumps	Located outside west of the Reactor Auxiliary Building, and next to the Reactor Building at elevation 19.5 feet MLW.	Vulnerable to flooding above elevation 19.5 feet MLW.
Component Cooling Water (CCW) Pumps	Located outside east of the Fuel Handling Building in the CCW equipment area.	Base elevation of equipment is 24 feet MLW. Vulnerable to flooding due to inundation and wave action above 24 feet MLW.
Component Cooling Water Heat Exchangers	Located outside east of the Fuel Handling Building on the CCW equipment area.	Base elevation of equipment is 24 feet MLW. Vulnerable to flooding due to inundation and wave action above 24 feet MLW.
Intake Cooling Water Pumps	Located on the west side of the plant site on the Intake Structure. Pump base elevation is 22 feet MLW.	Vulnerable to flood and wave action for elevations greater than 22 feet MLW.
Diesel Oil Storage Tanks (2)	Northeast of Diesel Generator Building, base elevation at 22.5 feet MLW.	Flood protection is provided up to 24.5 feet MLW.
Switchyard	Located west of the plant at elevation 18 feet MLW.	Vulnerable to inundation and wave action above elevation 18 feet MLW.

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TABLE 2-2 (Continued)

VULNERABLE STRUCTURES AND EQUIPMENT

Structure/Equipment	Location	Comment
Diesel Generators	Located in the Diesel Generator Building near the east side of Reactor Auxiliary Building. Operating floor elevation is 22.7 feet MLW.	Vulnerable to inundation above elevation 22.7 feet MLW.
4.16KV Switchgear	Switchgear 1A-2 and 1B-2 are located at the south end of the Turbine Building, floor elevation is 19.5 feet MLW.	Vulnerable to inundation above elevation 19.5 feet MLW.
	Switchgear 1A-3 and 1B-3 are located in the Reactor Auxiliary Building at elevation 43 feet MLW.	Not vulnerable to flooding.
	Switchgear 1AB is located in the Reactor Auxiliary Building at elevation 28.7 feet MLW.	Vulnerable to flooding above elevation 28.7 feet MLW.
480V Switchgear	Switchgear 1A-1, 1B-1 and AB are located on the south side of the Turbine Building at elevation 19.5 feet MLW.	Vulnerable to flooding above elevation 19.5 feet MLW.
	Switchgear 1A-2 and 1B-2 are located in the Reactor Auxiliary Building at elevation 43 feet MLW.	Not vulnerable to flooding.

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TABLE 2-2 (Continued)

VULNERABLE STRUCTURES AND EQUIPMENT

Structure/Equipment	Location	Comment
480V Motor Control Centers (MCC)	All 480V MCCs are located in the Reactor Auxiliary Building at elevation 43 feet MLW.	Not vulnerable to flooding.
Batteries	Batteries 1A and 1B are located on the east side of the Reactor Auxiliary Building at elevation 44.8 feet MLW.	Not vulnerable to flooding.
Battery Chargers	Battery chargers 1A and 1B are located on the east side of the Reactor Auxiliary Building at elevation 43 feet MLW.	Not vulnerable to flooding.
Control Room	Located in the north side of the Reactor Auxiliary Building at elevation 62 feet MLW.	Not vulnerable to flooding.
Remote Shutdown Panel	Located in the Reactor Auxiliary Building at elevation 43 feet MLW.	Not vulnerable to flooding.

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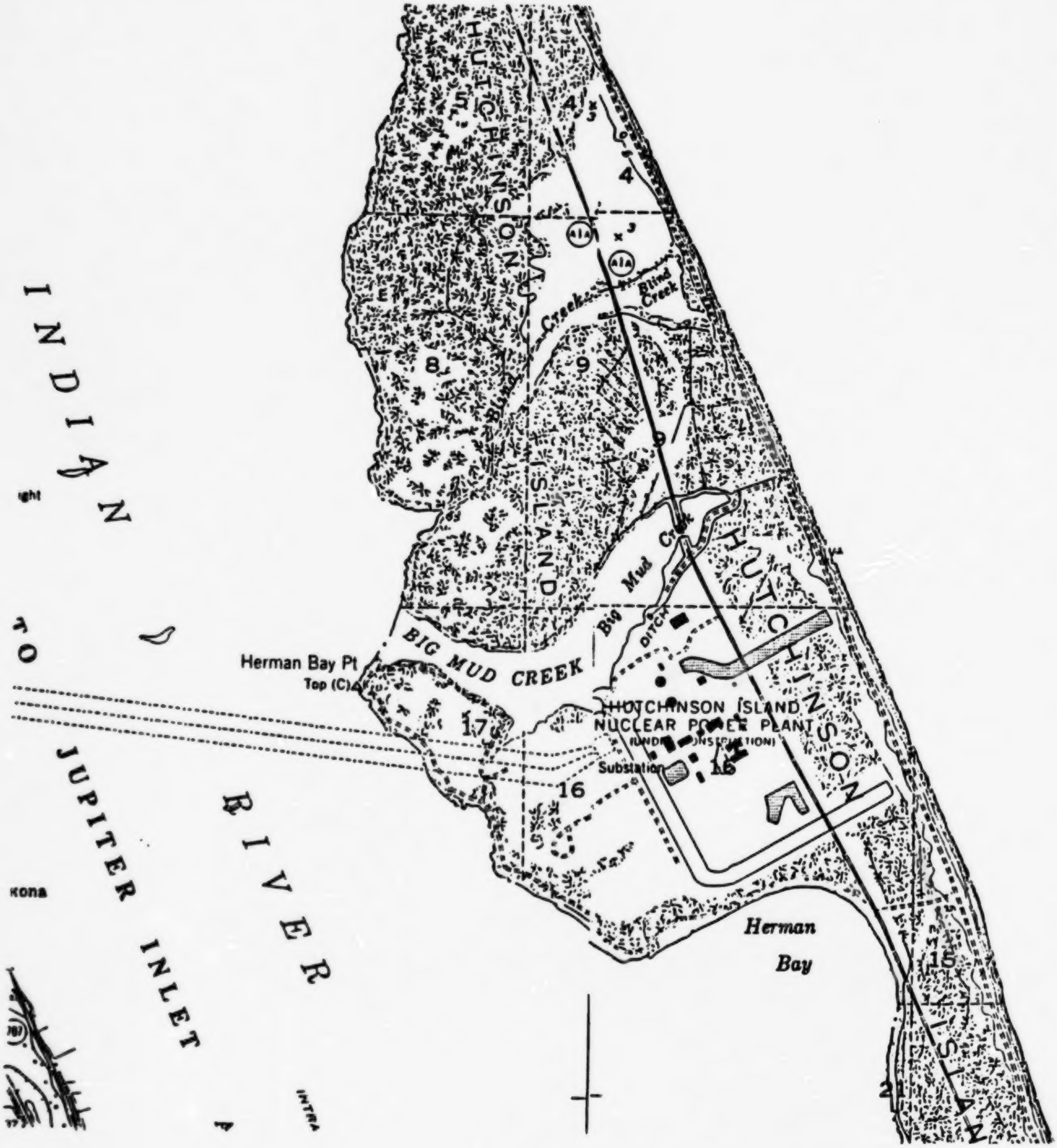
INDIAN RIVER
TO JUPITER INLET
HUTCHINSON ISLAND
HUTCHINSON ISLAND
Blind Creek
Blind Creek
Big Mud Creek
BIG MUD CREEK
Ditch
A1A
A1A
Herman Bay Pt
Top (C)
HUTCHINSON ISLAND NUCLEAR POWER PLANT
(UNDER CONSTRUCTION)
Substation
Herman Bay
4
4
5
8
9
9
15
16
16
17
21
ight
Kona
INTRA
707

FIGURE 2-1 ST. LUCIE AND SURROUNDING AREA
(FROM USGS TOPOGRAPHIC MAP)

0550

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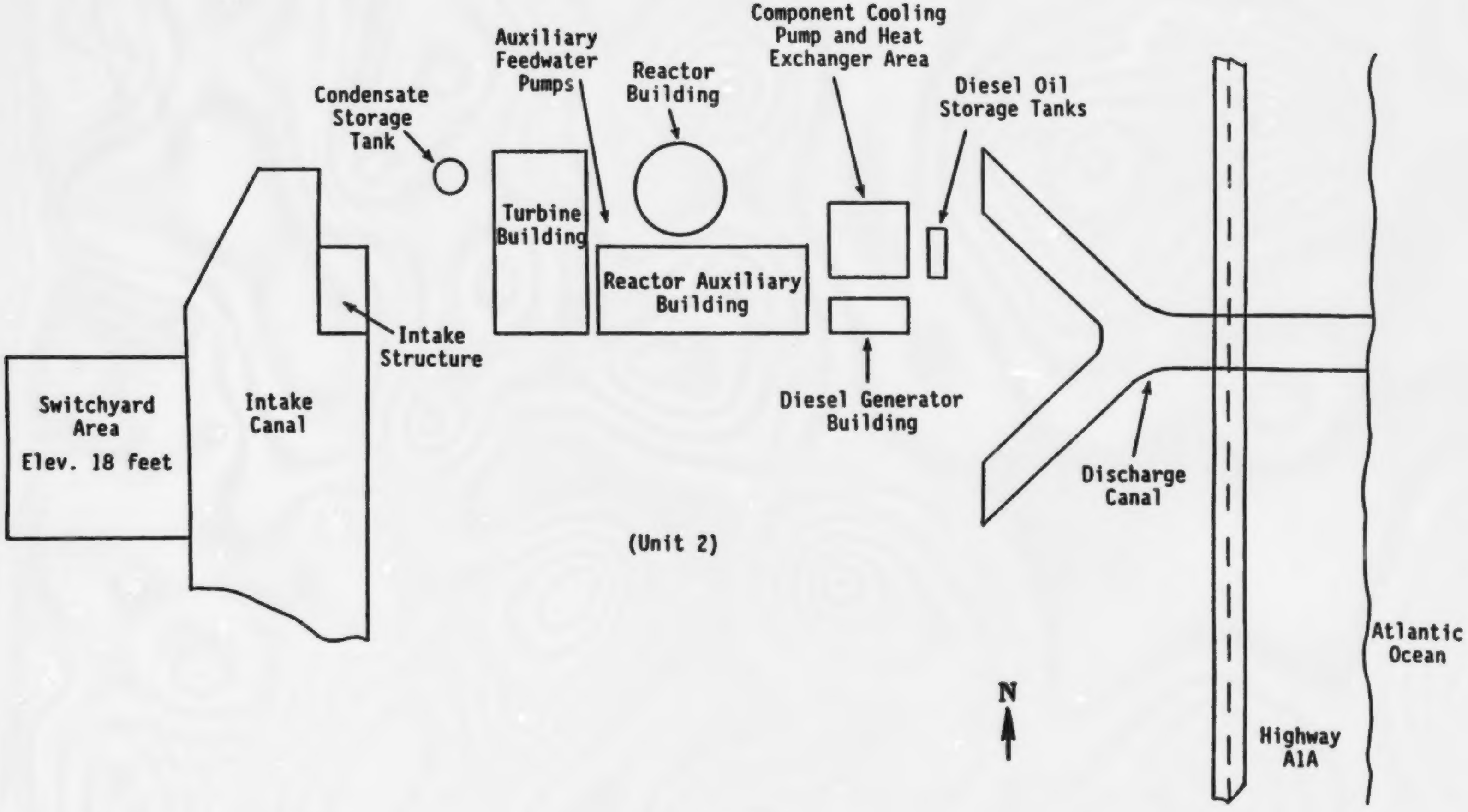
Component Cooling Pump and Heat Exchanger Area
Auxiliary Feedwater Pumps
Reactor Building
Diesel Oil Storage Tanks
Condensate Storage Tank
Turbine Building
Reactor Auxiliary Building
Intake Structure
Switchyard Area
Elev. 18 feet
Intake Canal
Diesel Generator Building
Discharge Canal
(Unit 2)
Atlantic Ocean
N
Highway A1A

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FIGURE 2-2 ST. LUCIE PLOT PLAN

3.0 FLOOD HAZARD ANALYSIS

Unlike other natural hazards such as earthquakes and extreme winds, external flood events may not present an adverse hazard to safety-related structures or equipment at a nuclear plant. Due to topographic or other physical conditions, flooding at a site may be precluded from occurring at all, or may have an extremely low frequency of occurrence. This is due in part to the fact that the design basis floods for nuclear plants are generally conservative. However, since the uncertainty in design floods was not considered, the likelihood of extreme floods and the contribution to plant risk is not known.

In the flood hazard assessment a series of steps was taken to evaluate the significant hazards that could impact a plant. In the first step the potential sources of flooding in the vicinity of the plant were identified. Next, a screening analysis was performed to establish for each source of flooding whether the plant site could be impacted. The results of the screening analysis for St. Lucie are described in Section 3.1.

For flood hazards which could adversely impact the plant, a simplified analysis was conducted to evaluate the probability distribution on the frequency of flooding the site. The hazard analysis methodology is described in Section 3.2. The results of the analysis are given in Section 3.3.

3.1 Screening Analysis

In the screening analysis, potential sources of flooding at St. Lucie were identified and evaluated. The objective of this first phase of the flood hazard assessment was to establish which flood events required direct consideration in the risk analysis. For those sources of flooding that do not adversely impact the plant, the basis for eliminating them from further consideration is provided.

The scope of the screening analysis generally involved simplified, conservative assessments to determine whether a particular source of flooding adversely impacted plant structures and equipment. If on this basis the integrity of plant components necessary for the removal of decay heat was not jeopardized for a given type of flooding, no further consideration was warranted.

Table 3-1 lists the potential sources of flooding in the vicinity of St. Lucie. These sources of flooding were identified based on information provided in the St. Lucie FSAR (Refs. 2 and 3). References 2 and 3 mention tsunami as a possible cause of flooding. However, the physical attributes of seismic events necessary to generate a sea wave do not exist in the Atlantic Ocean; therefore, this flood phenomena is readily discounted.

Hurricane Storm Surge

The greatest potential for flooding St. Lucie is due to storm surge and wave action resulting from a hurricane (Refs. 2 and 3). Historically, the largest observed tides occurred in 1949, which is believed to have produced a peak elevation of 8.5 feet MSL at Stuart and 11.3 feet MSL at Ft. Pierce. Table 2-1 summarized some of the large floods that have occurred in the vicinity of the plant.

For purposes of the screening analysis, an estimate of the frequency of flooding at plant grade was made. To do this the results of the flood insurance study performed for St. Lucie County, Florida (Ref. 6) were used. The flood insurance study provides an estimate of the frequency of flooding based on a joint probability analysis (Ref. 7). Table 3-2 shows the results of the flood frequency study for St. Lucie.

For purposes of the screening analysis, a Type I Extreme Value distribution and a Lognormal distribution were fitted to these data. From this it was estimated that the frequency of flooding to plant grade ranged from 1.0-6 to 1.0-5[1]. This estimate does not include the added effect of waves.

Based on this preliminary assessment that the frequency of flooding plant grade could exceed 1.0-6, it was concluded that a more comprehensive assessment of the frequency of on site flooding due to storm surge should be performed.

As part of the storm surge analysis reported in the FSAR, an evaluation was conducted to assess the amount of erosion that would occur near the plant due to the PMH. Based on a conservative analysis, it was concluded that extensive erosion could occur in the vicinity of the plant. However, considerable margin remains in terms of the reserve distance between plant structures and the estimated advance of erosion. Due to the conservatism in the erosion analysis and the fact that the occurrence of extreme erosion is conditional on the occurrence of a PMH size storm (i.e., a frequency of 1.0-6 to 1.0-5 per year), a detailed probabilistic evaluation of erosion is not performed since its frequency of occurrence will be less than 1.0-6 to 1.0-5.

Local Precipitation

St. Lucie is located along the southeastern part of the Florida peninsula. The climate is subtropical marine, characterized by long summers with abundant rainfall. Rainfall occurs principally in the months from April to October. Intense local precipitation often accompanies tropical storms and hurricanes.

The average annual rainfall amount recorded at West Palm Beach is 61.54 inches (Ref. 10). For the months of April to October the average rainfall varies from 3.49 inches in May to 9.45 inches in September. The maximum rainfall recorded in a 24-hour period was 15.23 inches which occurred in April 1942. According to Reference 11, the Probable Maximum Precipitation for Florida is approximately 35 inches for a 24-hour duration for a 10 square mile area. According to the Unit 1 FSAR the 24-hour PMP is 32 inches, whereas the Unit 2 FSAR suggests the PMP is 47.1 inches.

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[1] 1.0-5 = 1.0×10^{-5}

The St. Lucie plant is constructed on an engineered fill whose elevation is 18 feet MLW. The terrain in the immediate vicinity of the plant has an elevation ranging from 2 to 5 feet MLW. Therefore, the plant is approximately 13 to 16 feet above the local terrain.

The area of the plant at elevation 18 feet MLW is relatively small and given the local topography, rainfall on site tends to runoff to the low lying areas. Due to the difference in elevation between the plant and the local terrain, runoff from the surrounding area does not contribute to flooding on site. Therefore, direct precipitation on site will pose the principal contribution to possible flooding.

According to the FSAR (Ref. 2) the plant site and building roofs have been designed to withstand the affects of the PMP. With the exception of the Reactor Building, no water build-up on the roofs of safety-related structures exceeds 2 inches. Buildings are provided with sloping roofs, exterior leaders to the storm drainage system and little or not parapet walls. The Reactor Building has a domed roof with a 1.5 foot parapet wall. Drainage is provided by exterior leaders from the parapet to the storm drainage system.

The results of the site analysis indicate that the capacity of the site drainage system may be exceeded, particularly in cases when rainfall is accompanied by wave-runup during a tropical storm or hurricane (Refs. 2 and 3). However, the FSAR for both units concludes that the degree of on-site ponding is self limiting due to the fact that the crest of local catch basins is generally 18 feet MLW. However, the plant access road has a crown elevation of 19 feet MLW. Since the lowest elevation of doors to critical structures is 19.5 feet MLW, water captured in localized basins will overflow into low lying areas, away from plant structures. The FSAR also notes that the site drainage system is not connected to any Class I structure, with the exception of the roof leaders.

The roofs of buildings housing safety-related equipment have been designed to handle the rainfall associated with a PMP. The roof leaders were designed for a maximum rainfall intensity of 6 inches per hour. According to the FSAR (Ref. 2) if rainfall intensities exceed 6 inches per hour, or if the roof leaders are partially clogged, water build-up cannot exceed 2 inches. For the Reactor Building, if the roof leaders become clogged, water could accumulate at the parapet wall. However, due to the shape of the dome roof any accumulation is concentrated over a small area. In effect a wedge of water is formed such that the maximum water depth would be 18 inches at the wall itself, decreasing to zero due to the dome shape. The average depth of water would be approximately 11.4 inches for a width of 74 inches.

For purposes of the screening assessment it was assumed that the leader drains were completely blocked, resulting in the maximum accumulation of water on the Reactor Building roof. This increased load was compared to the design load cases and allowable material strengths as defined in Section 3 of the FSAR (Ref. 2). The load due to the water accumulation represents approximately a 2 percent increase in load case (i) which is the critical load for the dome. Since there is an available margin of at least 50 percent in the design strength of the Reactor Building, it is concluded that the roof load due to water accumulation represents an insignificant increase in the design load:

From a review of the natural drainage and previous experience at the plant, it is concluded that ponding in the vicinity of plant structures would not affect Category I structures and essential equipment. This is based on input provided by the FSAR which indicates that if the storm 0556 drainage capacity is exceeded, ponding in low lying areas is limited to elevations for at least one-half foot below the lowest door openings. Also, any increased load on the Reactor Building roof due to water accumulation was determined to be small. As a result, a more in depth hazard assessment for local precipitation is not warranted.

3.2 Hazard Assessment

Based on the results of the screening analysis, it was concluded that a probabilistic hazard assessment for the effects of hurricane storm surge and wave action should be performed. The possibility of flooding the St. Lucie site due to storm surge was considered in the design basis of the plant. Specifically, the plant was constructed such that safety-related structures are situated above the design basis flood level.

3.2.1 Methodology

To evaluate the frequency of storm surge, the joint probability method of analysis is generally used (Refs. 6-9). This approach consists of three basic parts. The first is a numerical model which is used to predict the time variation of storm surge for a given set of hurricane input parameters (e.g., central pressure, radius to maximum windspeed, translational speed, etc.). Next, data on historic hurricane occurrences are used to estimate statistical distributions of the key parameters in the analysis. The third step involves running the numerical storm surge model for all possible joint, random combinations of the hurricane parameters. In applications of the joint probability method, model and parameter (e.g., mean of the distribution on the central pressure) uncertainties are not incorporated in the analysis. Therefore, a single flood hazard curve is estimated.

The joint probability method was used to estimate the frequency of flooding in the St. Lucie area (Ref. 6). Performed as part of a flood insurance study, flood elevations are reported for frequencies ranging from 2.0-3 to 1.0-1 per year. These results are reported in Table 3-2.

As indicated in the screening analysis, the frequency of much more infrequent events must be estimated in order to evaluate the likelihood of flooding the plant site.

There is considerable difference of opinion regarding which, if any, procedure should be used to evaluate the frequency of extreme floods. Furthermore, there is a limited body of experience in the probabilistic assessment of extreme flood hazard phenomena the size of which impact nuclear stations since they are generally designed to conservative, deterministic standards. In the flood hazard analysis for TAP A-45, a simplified approach is used that provides estimates of the frequency of floods that impact the plant and assesses the uncertainty in the frequency distribution.

It is beyond the scope of this project to conduct a comprehensive joint probability analysis, including model uncertainty, in order to estimate the probability distribution on the frequency of extreme flood elevations. Therefore, a statistical approach is used to estimate the flood frequencies. The methodology to estimate the probability distribution on the frequency of flooding is discussed below.

In using a statistical approach to evaluate flood hazard curves, there are a number of factors that contribute to the uncertainty in the flood frequency estimates. As a result, there is a probability distribution on the estimate of the frequency of exceeding a given flood level. The uncertainty in the assessment of a flood hazard curve is a function of the following factors:

- Choice of the statistical frequency distribution
- Statistical uncertainty in the frequency distribution parameter estimates.

To characterize the uncertainty in each part of the analysis, alternative hypotheses are considered. A subjective probability is assigned to each hypothesis, which expresses the degree-of-belief that it represents the true alternative. For example, three alternative statistical frequency distributions could be selected as credible alternatives to characterize the uncertainty in defining the mean frequency of exceedance. Subjective probability weights are then assigned to each model, expressing the credibility given to the hypothesis that any model is the "true" frequency distribution.

In this analysis, statistical distributions are fit to the results of the joint probability analysis (Table 3-2). In effect, the frequency estimates provided by the joint probability study are used as data to determine the parameters of a statistical distribution. Once the parameters are determined, the results of the joint probability analysis can be extrapolated to estimate the frequency of flooding plant structures. It is assumed that this approach provides a median estimate of the frequency of exceeding flood elevations.

An important part of estimating the frequency of flooding is the selection of the statistical distributions. In this analysis three models are used to account for the uncertainty in the statistical distribution.

Due to limited data, uncertainty exists in the estimate of the parameters of a statistical frequency distribution, resulting in a corresponding uncertainty in the estimate of the frequency of exceeding a given elevation. Based on the statistical uncertainty, a family of flood hazard curves (i.e., frequency of exceedance curves) are estimated. The amount of uncertainty in the frequency estimates is a function of the fit of a given frequency distribution to the data.

The elements of the uncertainty analysis can be displayed in a logic diagram as shown in Figure 3-1. The logic diagram illustrates for each source of uncertainty (i.e., choice of statistical frequency distribution and statistical uncertainty), the alternative hypotheses and the subjective probability weights assigned to each model. Along a branch of the diagram, a frequency distribution is defined, based on the models and parameters indicated. The product of the individual probabilities gives the subjective weight assigned to that result.

The final product is a discrete probability distribution on the frequency of flooding, composed of individual elements, $\nu(L)_k$, $(p_i q_j)$, where $\nu(L)_k$ is the kth estimate of the frequency distribution on stillwater elevation, L, and $(p_i q_j)$ is the subjective probability that this result is the "true" hazard curve. The family of hazard curves expresses the uncertainty in estimating the frequency of exceedance. From the resultant probability distribution on frequency, a mean (or weighted average) hazard curve can be determined.

To estimate the frequency distribution on peak surge elevation, the following statistical models were selected,

- Lognormal
- Gumbel or Extreme Value Type I
- Log Pearson Type III

Based on the fit of each statistical model to the data (e.g., tidal data or the joint probability analysis results), a probability distribution on the model parameters was obtained. In the case of the Lognormal model, for example, a joint normal distribution on the mean and standard deviation of the logarithms of the data was obtained, $f(m,\sigma)$. By discretizing the set of m and σ values, a family of hazard curves was derived for the Lognormal model that express the statistical variability in the estimate of the frequency of exceedance. The probability weight assigned to each curve, q_i, is equal to,

$$q_i = \int_{m}^{m+\Delta m} \int_{\sigma}^{\sigma+\Delta\sigma} f(m,\sigma)\,dm\,d\sigma \qquad (3.1)$$

This theme was repeated for the other models to determine their statistical variability.

0560

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In addition to a storm surge, a hurricane will cause wave action and runup at the plant. The wave height will vary according to location and changes in topography. For example, due to the depth of water in the discharge canal, high waves will occur. However, in proximity to plant structures, the depth of water will be less, and thus the wave heights will not be as great. In this analysis the wave height is added to the final stillwater flood hazard curves to estimate the total flood hazard level. This approach to predict the final flood elevation is a simplification of a rather complex process. It is, however, consistent with the approach used in flood insurance studies and in the St. Lucie FSAR.

3.2.2 Results

The quantification of the flood hazard at St. Lucie was performed using the steps outlined in Figure 3-1. A total of 15 flood hazard curves that express the uncertainty in estimating the frequency of exceedance of peak flood elevation were developed; 3 (statistical frequency distributions) x 5 (hazard curves due to statistical uncertainty).

Subjective probability weights were assigned to each statistical frequency distribution to reflect the belief that any one of them is the true model. The following weights were assigned:

Statistical Distribution	Subjective Probability
Lognormal	0.34
Extreme Value	0.33
Log Pearson	0.33

These weights reflect an indifference to the choice of the statistical frequency distribution.

The 15 hazard curves derived in the flood frequency analysis are aggregated to obtain 5 curves which have equal weight. Table 3-3 and Figure 3-2 show the results of the analysis. Also given in Table 3-3 is the mean (or weighted) average of the 5 curves. The mean frequency of stillwater surge elevations exceeding plant grade of 19 feet MLW is approximately 1.4-5 per year. The frequency of exceeding elevation 19.5 feet MLW is approximately 8.09-6.

The large uncertainty in the hazard curves (see Fig. 3-2) is a function of the large statistical variability in estimating the parameters of each statistical model particularly at elevations near plant grade. As a result of the large uncertainty, the mean frequency estimates are skewed toward the higher probability levels of the distribution.

Wave Action

In the event of a major hurricane, waves will travel up the discharge canal (Fig. 2-1) and runup the embankment which is the boundary of the nuclear island. For stillwater elevations above 18 feet MLW, wave action in the immediate vicinity of the plant will be small due to the fact that the depth of water is shallow. For example, a stillwater flood level of 19 feet results in water only 1 foot deep at the edge of the embankment whereas in the surrounding area the water will be 9 to 17 feet deep.

The design basis of the plant included a stillwater elevation of 16.22 feet MLW and waves at the upstream end of the discharge canal of approximately 10.3 feet. In addition, runup along the banks of the discharge canal is expected. A flood barrier with a top elevation of 22 feet MLW is designed to mitigate the possibility of runup into areas where safety-related components are located.

On the nuclear island the effect of wave action and runup is difficult to predict because of the arrangement of buildings and the

0562

floodwall which tend to break the transmission of waves throughout the plant. Also, the embankment around the plant will tend to insure that waves break away from plant structures.

In order to estimate wave height and runup levels, the simplified procedures in the Shore Protection Manual (Ref. 12) were used. For purposes of estimating the runup on site, the embankment slope conditions at the nose of the discharge canal were used. Table 3-4 summarizes the wave height and runup elevation estimates. The wave heights reported in Table 3-4 correspond to waves at the nuclear island embankment, and not the deep water waves generated in the discharge canal.

From the results of the wave height and runup analysis, a stillwater elevation of 19 feet MLW with accompanying runup results in a total flood elevation of 21 feet MLW. A flood of this intensity would produce a stillwater level to plant grade near safety-related structures and runup due to wave action of 2 feet. This would exceed flood protection levels of 19.5 feet MLW, resulting in leakage into plant buildings.

The true extent of flooding that would occur into a building is a function of a number of factors. These include the duration of the storm, the effect of the arrangement of buildings on site and the precautions taken prior to the arrival of the storm. Since the stillwater flood levels predicted are peak values, they do not represent a sustained level of inundation that would be sustained. Consequently, use of peak flood levels is conservative.

TABLE 3-1

SCREENING ANALYSIS SUMMARY OF EXTERNAL FLOOD HAZARDS AT ST. LUCIE

Source of Flooding	Type of Flooding	Status
Atlantic Ocean	Hurricane storm surge and wave action.	Hazard analysis required.
Local Precipitation	Summer thunder storms and precipitation associated with tropical storms and hurricanes.	Hazard analysis not required.

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0564

TABLE 3-2

FLOOD INSURANCE STUDY RESULTS
(Ref. 6)

Frequency	Stillwater Flood Elevation (feet MLW)
1.0-1	5.5
2.0-2	7.5
1.0-2	8.4
2.0-3	10.2

TABLE 3-3

STILLWATER FLOOD HAZARD CURVES

Curve No.	Subjective Probability	Elevation (feet MLW)						
		16	18	19.5	20	21	22	24
1	0.20	3.39-6	6.43-7	1.81-7	1.19-7	5.30-8	2.40-8	5.30-9
2	0.20	1.20-5	2.50-6	8.17-7	5.67-7	2.77-7	1.38-7	3.22-8
3	0.20	2.73-5	6.57-6	2.28-6	1.62-6	8.36-7	4.39-7	1.30-7
4	0.20	5.57-5	1.62-5	6.71-6	4.95-6	2.67-6	1.46-6	4.42-7
5	0.20	1.94-4	6.56-5	3.04-5	2.39-5	1.50-5	9.41-6	3.85-6
	Mean Hazard	5.85-5	1.83-5	8.09-6	6.24-6	3.77-6	2.29-6	8.92-7

TABLE 3-4

WAVE HEIGHT AT THE NOSE OF THE DISCHARGE CANAL AND DIESEL OIL STORAGE TANKS

Stillwater Flood Elevation (feet MLW)	Wave Height (feet)	Runup (feet)	Runup Elevation (feet MLW)
16	1.1	0.8	16.8
18	2.2	1.6	19.6
19	2.7	2.0	21.0
20	3.3	2.3	22.3
22	4.3	3.1	25.1

0567

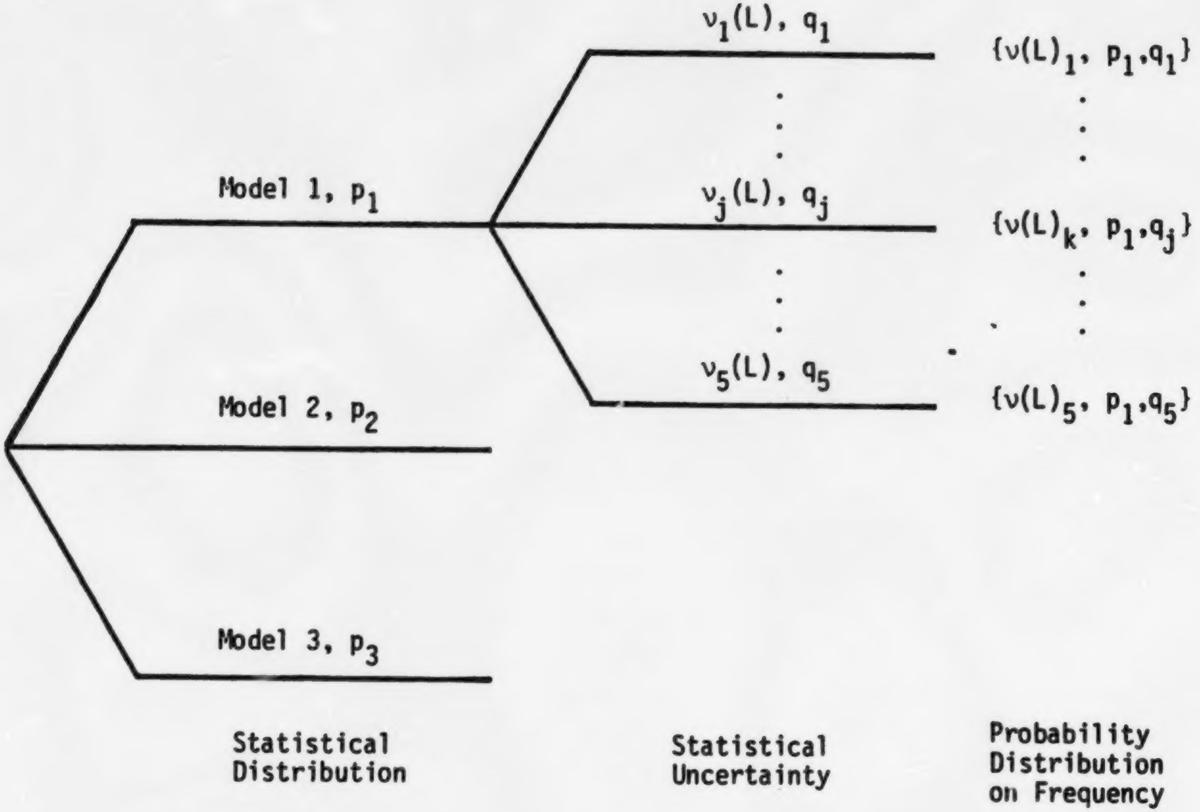
Model 1, p_1
Model 2, p_2
Model 3, p_3
$\nu_1(L)$, q_1
$\nu_j(L)$, q_j
$\nu_5(L)$, q_5
$\{\nu(L)_1, p_1,q_1\}$
$\{\nu(L)_k, p_1,q_j\}$
$\{\nu(L)_5, p_1,q_5\}$
Statistical Distribution
Statistical Uncertainty
Probability Distribution on Frequency

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FIGURE 3-1 LOGIC DIAGRAM ILLUSTRATING THE FLOOD HAZARD UNCERTAINTY ANALYSIS

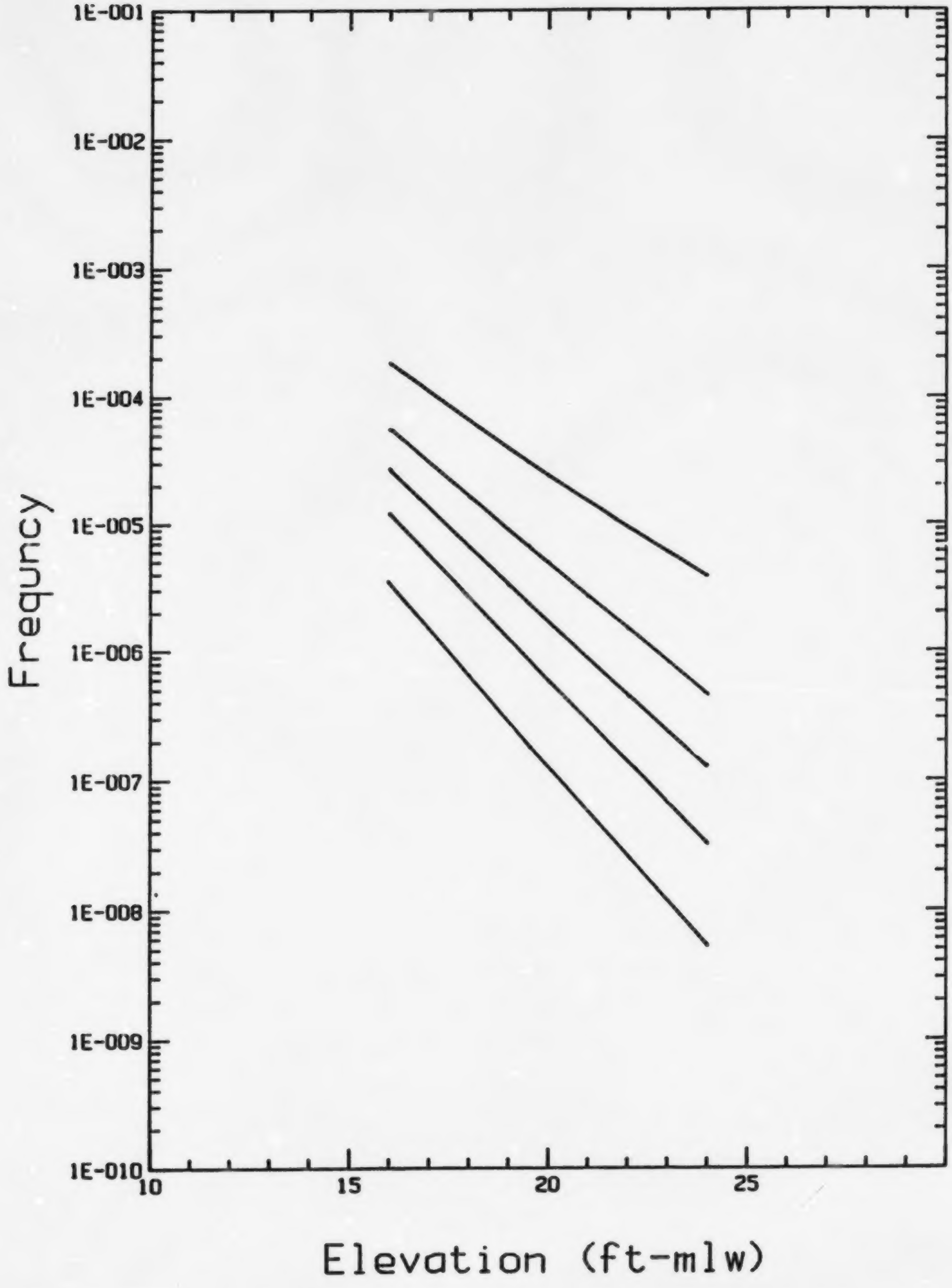

FIGURE 3-2 ST. LUCIE PLANT FLOOD HAZARD CURVES, STILLWATER LEVEL

0569

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4.0 FRAGILITY ANALYSIS

In this section fragility data for structures and equipment items vulnerable to the effects of flooding are developed. As indicated in Section 1.0, a conservative approach was used in developing capacities of structures and equipment to resist the effects of flooding. If it is found in the systems analysis that external flooding is a significant contributor relative to other external and internal initiators, a more detailed look at components which dominate the flood contribution may be required.

In Section 4.1 the methodology to evaluate the fragility of structures and equipment items is described. For the components listed in Table 2-2 which are vulnerable to the effects of external floods, fragility data are reported in Section 4.2.

4.1 Methodology

In this section the approach used to evaluate fragility data for structures and equipment items is described. The fragility assessment for plant structures is described first.

Structures

To estimate the fraction of failure of plant structures due to flood loads, an approach similar to that used in seismic analysis was adapted in which the capacity of a structure is expressed as a lognormal distribution. The parameters of the model that define the fraction of failure due to flood forces are:

- Median capacity defined in terms of flood elevation or depth

- Logarithmic standard deviation for capacity due to randomness, β_r

- Logarithmic standard deviation of the median capacity due to uncertainty, β_u

The median is a best estimate of the capacity of a structure. The fraction failure at this level is 0.50.

The two logarithmic standard deviations, β_r and β_u characterize the various sources of variability that exist in predicting structural failure. The first term, β_r, corresponds to the inherent randomness in the capacity of a structure to resist loads. An example is the natural variability in material properties. The second term, β_u, expresses the uncertainty in estimating the median capacity of a structure, or the location about which the fragility curve is centered.

The fragility of a structure is defined by the lognormal distribution which is expressed as:

$$F(L) = \Phi\left[\frac{\ln(1/\bar{L})}{\beta_r}\right] \quad (4.1)$$

where: $\Phi[\ ]$ = the standard cumulative normal distribution
$F(L)$ = fraction of failure evaluated at flood depth (or elevation), L
$\bar{L}$ = median capacity of the structure

In the above expression $\bar{L}$ is also uncertain, as defined by β_u. Thus, a family of fragility curves can be defined for a set of possible estimates of $\bar{L}$.

A number of factors contribute to the capacity of a concrete structure for the effects of flood loads. They include:

0571

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- hydrostatic and dynamic wave forces
- pattern of cracking and failure
- failure level of a structure.

Fragility parameters are based on design information when possible or simplified calculations.

To access the capacity of plant structures, the fragility evaluation is made in terms of the flood elevation which includes stillwater flood levels and corresponding wave action at the plant site.

Equipment Items

Failure of components is assumed to occur due to submergence. Therefore, failure of a component occurs when a critical flood depth is reached. In the case of inundation of the plant site due to storm surge, the fragility of equipment must ultimately be expressed in terms of the flood elevation on site inundation. For example, the fragility of components located in rooms below plant grade must be expressed in terms of a flood elevation, and not the depth of water in a room needed to submerge a particular component. In this way, the characterization of the hazard and fragility parts of the risk assessment are compatible.

To evaluate the fragility of equipment items exposed to external flooding, the following steps were considered:

- Estimate the depth of submergence necessary to fail a component in a room
- Identify flood protection devices (if any)
- Determine modes of flooding a room or area (i.e., propagation through doorways, down stairs, etc.)

0572

- Transform local fragility values or critical depth estimates to a global flood hazard characterization

For each component vulnerable to flooding these steps were taken and fragility data estimated.

4.2 Results

Structures

The structures at St. Lucie have been designed for a number of extreme load conditions. Section 3.8.1.5 of the FSAR (Ref. 2) defines the load combinations and load factors that were used. With respect to the design of plant structures to resist the effects of external flooding, a flood level of 16.2 feet MLW was considered.

Below grade only the Reactor Auxiliary Building houses equipment necessary which are important to decay heat removal. If cracking of walls below grade occurs, flooding of the building interior would result. According to Section 3.8 of the FSAR, the critical sections of the exterior walls of the Reactor Auxiliary Building have a factor of safety of at least 1.8 for shear and 2.50 for moment at the base of the wall. For a flood elevation of 19 feet MLW, the total hydrostatic load increases approximately 23 percent, reducing the minimum margin to 1.45.

According to the FSAR, the Intake Structure is also impacted by extreme hydrostatic loads. This is due in part to the depth of water in the intake canal. In Section 3.8 of the FSAR, results of structural calculations are compared to the ultimate capacity of critical elements. The factor of safety for the governing load combination varies from 1.12 to 1.86.

Based on this information, fragility estimates for the Reactor Auxiliary Building and Intake Structure were estimated. The parameters of this distribution are given in Table 3-1. A fragility estimate for the Reactor Building is not included since it was designed to even more severe flood loads than the Reactor Auxiliary Building. A flood level of 21 feet MLW was used, rather than 16 feet MLW.

Equipment

Table 4-2 summarizes the fragility data for safety-related equipment. The fragility estimates are based on the results of the plant walkdown and a review of plant drawings. It was assumed as part of the fragility analysis for equipment that the three foot floodwall provided only marginal protection against flooding. This is based on the most recent information that indicates stop logs will not be available to close openings in the floodwalls. On the otherhand the flood walls and general arrangement of plant structures will serve as a means to limit the direct impact of wave runup to building openings.

For equipment located on the ground floor of buildings it was assumed, based on observations during the plant visit, that there is a 0.50 foot margin between the base level and the level that would cause failure. In addition, to incorporate the effects of runup into the building, the stillwater elevation plus the additional runup must reach an elevation 1.0 above the critical flood depth of the component. This simplified approach accounted for the leakage through exterior doors and the fact that the runup is not a sustained inundation level.

For example, a component which is located just inside a doorway on the ground floor level of 19.5 feet MLW has a critical elevation of 20 feet MLW. In this case the critical flood has a stillwater elevation of 19.0 feet MLW and a calculated runup of 2 feet.

0574

TABLE 4-1

STRUCTURE FRAGILITY ESTIMATES

Structure	Median		β_r	β_u	β_c
	(ft MLW)	(ft Depth)			
Reactor Auxiliary Building	30	25*	0.05	0.25	0.25
Intake Structure	16.8*	47.8	0.05	0.25	0.25

* Use the value indicated as the median of the lognormal distribution.

TABLE 4-2

EQUIPMENT FRAGILITY ESTIMATES

Component	Critical* Elevation (feet MLW)
Switchyard	15.2*
Diesel Generators	23.2
Diesel Oil Storage Tanks	24.5
Containment Spray Pumps	Not Vulnerable
HPSI Pumps	Not Vulnerable
LPSI Pumps	Not Vulnerable
Shutdown Heat Exchanger	20.0
Auxiliary Feedwater Pumps	19.0*
Component Cooling Water Heat Exchangers	22.0*
Component Cooling Water Pumps	22.0*
Batteries	
- Reactor Auxiliary Building	Not Vulnerable
Battery Chargers	Not Vulnerable
480V Switchgear	
- 1A-1	19.0*
- 1B-1	19.0*
- AB	29.2

* Elevations with an asterisk include the effects of wave action and runup.

0576

TABLE 4-2

EQUIPMENT FRAGILITY ESTIMATES
(CONTINUED)

Component	Critical* Elevation (feet MLW)
480V Motor Control Centers	Not Vulnerable
4.16V kV Bus	
- 1A-2, 1B-2	19.0*
- 1A-3, 1B-3	Not Vulnerable
- 1AB	> 22
Remote Shutdown Panel	Not Vulnerable
Control Room	Not Vulnerable
Intake Cooling Water Pumps	22

* Elevations with an asterisk include the effects of wave action and runup.

5.0 SYSTEM ANALYSIS

External flooding can potentially cause failure of the safety systems in the plant. In this section the response of the plant to the safety system failures is discussed. A procedure is given that is used in Section 6.0 for quantifying risk.

The methodology used to perform the systems analysis is given in Section 5.1. The results are given in Section 5.2.

5.1 Methodology

The principal cause of flooding at St. Lucie that could potentially cause failure of the plant's safety systems is storm surge and wave action due to tropical storms and hurricanes.

To determine the contribution (if any) that tropical storms and hurricanes can have to core melt probability, the following steps are performed:

(1) The flood level that will fail sufficient safety systems resulting in core melt is determined.

(2) The frequency of an external flooding event that will result in the above flood level is determined.

(3) The first two steps will given the probability of core melt due to external flooding if no actions (recovery or otherwise) are taken by plant personnel. This gives an upper bound result. However, at St. Lucie procedures exist in the event the external flooding occurs. These procedures taken by plant personnel are incorporated into the analyses.

5.2 Results

The results for external flooding systems analysis are given in the following two subsections.

5.2.1 Component/System Failures Due to External Flooding

The systems that will fail as a result of the failure of the various vulnerable equipment/structures are shown in Table 5-1. The critical flood elevation (including wave action) is also given for the various components. The vulnerable equipment/structure listed in the left column are obtained from Table 2-1. The critical stillwater elevations are obtained from Table 4-2. In determining which systems will fail it was assumed that simultaneous loss of offsite power (LOSP) occurs. Each systems failure is discussed below.

(1) Switchyard

Flooding of the switchyard will result in the loss of offsite power (LOSP)

(2) HPIS/LPIS/HPRS/LPRS due to flooding of the diesel generator room

Flooding of the diesel generator room will only fail these systems since there is an assumed simultaneous loss of offsite power (LOSP) during the external flooding event. Both early (HPIS/LPIS) and later (HPRS/LPRS) emergency coolant injection and recirculation systems will fail. The AFWS turbine driven train will also fail due to battery depletion or it can fail randomly.

(3) HPIS/LPIS/LPRS/HPRS due to flooding the diesel oil storage tanks

Since a LOSP occurs, the diesel generators will fail assuming the diesel fuel oil tanks fail once each diesel generator has exhausted its own day tank capacity in approximately 8 hours. At that time the emergency coolant injection (ECI) systems HPIS and LPIS would fail. The AFWS would then also fail due to battery depletion in another 8 hours.

TABLE 5-1

SYSTEMS FAILED AS A RESULT OF EQUIPMENT/STRUCTURE EXTERNAL FLOODING AT SL. LUCIE

	Vulnerable Equipment/Structure Failed	Critical Elevation(1) (Feet-MLW)	Safety Systems(s) Failed
1.	Switchyard	15.2*	LOSP
2.	Diesel Generators	23.2	HPIS/LPIS/HPRS/LPRS(2)
3.	Diesel Oil Storage Tanks	24.5	HPIS/LPIS/HPRS/LPRS(2)
4.	Shutdown Heat Exchangers	20.0	LPRS(3)
5.	Auxiliary Feedwater Pumps	19.0*	AFWS
6.	Component Cooling Water Heat Exchangers	22.0*	HPIS/HPRS/LPIS/LPRS/CSIS/CSRS
7.	Component Cooling Water Pumps	27.0*	HPIS/HPRS/LPIS/LPRS/CSIS/CSRS
8.	480V Switchgear		
	1A-1	19.0*	NONE
	1B-1	19.0*	NONE
	AB	29.2*	NONE
9.	4.16V KV BUS		
	1A-2, 1B-2	19.0*	LOSP
	1AB	22	LOSP (HPIS pump 1C, CCW pump 1C, and ICW pump 1C)
10.	Intake Cooling Water Pumps	22	ICWS and all safety systems except AFWS and the diesel generators
11.	Containment Spray Pumps	-	Not Vulnerable
12.	HPSI Pumps	-	Not Vulnerable
13.	LPSI Pumps	-	Not Vulnerable
14.	Batteries		
	Reactor Aux. Building	-	Not Vulnerable
15.	Battery Chargers	-	Not Vulnerable

---

(1) Elevations with an asterick include the effects of wave action & run up

(2) The AFWS will also fail due to battery depletion

(3) Heat removal mode of LPIS

* Elevations with an asterik include the effects of wave action and runup.

(4) Shutdown Heat Exchanger

Flooding of the shutdown heat exchangers would fail the RHR system which would be used as a backup to HPIS in the LPIS mode.

(5) AFWS due to flooding of AFWS pumps

The flooding of the AFWS pumps will fail the AFWS. This will cause the loss of secondary cooling for the plant. The PCS will fail due to the assumed simultaneous loss of offsite power.

(6) HPIS/HPRS/LPIS/CSIS/CSRS due to flooding of the component cooling pumps
&(7) and/or heat exchangers

The component cooling pumps and heat exchangers provide cooling water to all the emergency coolant injection system pumps and the containment spray pumps. This would fail all the safety systems except the AFWS turbine trains.

(8) 480V Switchgear 1A-2, 1B-2, 1AB

Does not fail any shutdown safety systems

(9) 4.16KV Bus 1A-2, 1B-2, 1AB

Flooding of these buses results in LOSP.

(10) Intake Cooling Water Pumps

The flooding of the intake cooling water pumps will fail the intake cooling water systems. This would fail all the safety systems except the AFWS turbine train and the diesel generators.

(11) - (15) Systems not vulnerable to flooding

5.2.2 External Flooding Event Tree

A functional event tree that models the response of St. Lucie to external floods is shown in Figure 5-1. This tree assumes that a simultaneous LOSP occurs at the time of the external flooding. In Boolean equation form, the functions are systemically defined as the following:

Secondary Cooling = AFWS

where, AFWS = Auxiliary Feedwater System

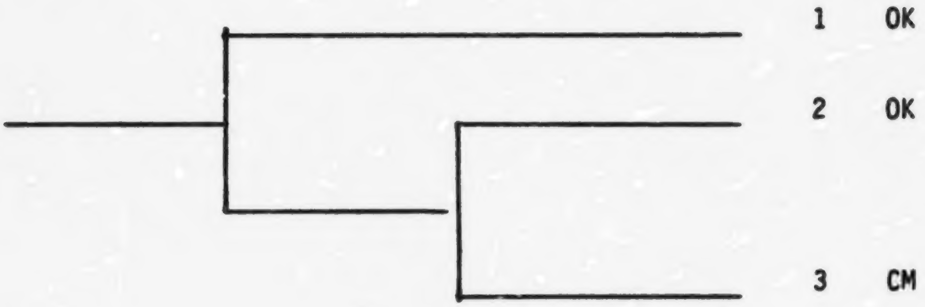
ECI = HPIS * LPIS + HPRS * LPRS

where, ECI = Emergency Coolant Injection
HPIS = High Pressure Injection System
LPIS = Low Pressure Injection System
HPRS = High Pressure Recirculation System
LPRS = Low Pressure Recirculation System

Under each function in the event tree in Figure 5-2 the support systems that will cause the functions to fail are listed. The loss of either the DGs or the diesel oil storage tanks will fail both functions. The loss of the AFWS will fail secondary cooling since the PCS fails due to LOSP. The ICW and the CCW systems provide cooling to the ECI system pumps. Only system failures that will cause the complete failure of each function are listed in Figure 5-1.

0582

Function	Secondary Cooling (SEC)	Emergency Coolant Injection (ECI)	Sequence Number	Core Status
Vulnerable Support Systems	DGs**, DOST**, AFWS	DGs, DOST CCW, ICW		



CCW = Component cooling water system (pumps and/or heat exchangers)
ICW = Intake cooling water system pumps
DGs = Diesel Generators
DOST = Diesel oil storage tanks
AFWS = Auxiliary feedwater system

** The AFWS can fail either randomly or due to battery depletion in eight hours after the DGs fail.

FIGURE 5-1. WIND/TORNADO FUNCTIONAL EVENT TREE FOR ST. LUCIE

6.0 RISK QUANTIFICATION

6.1 Methodology

In this section the core melt frequency contribution at St. Lucie due to external flooding is determined. This is done by using the system failure information and functional event tree developed in Section 5.0.

As can be seen from the functional event tree given in Figure 5-1, and from the analysis of the systems that will fail due to an external flood in Table 5-1 that there are two flood levels are important: 19 feet MLW and 22-feet MLW.

At 19-feet MLW the auxiliary feedwater systems will fail. This will result in the loss of the secondary cooling function. The emergency AC and high pressure injection systems are still available to provide the ECI function via feed-and-bleed (F&B). The random failure of feed-and-bleed is conservatively estimated to be 1E-1 at St. Lucie. Therefore, the core melt frequency becomes: (probability the water level reaches 19-feet MLW but does not exceed 22 feet) x (failure probability of F&B) = 9.21E-6 x 1E-1 = 9.21E-7.

At 22-feet MLW the intake cooling water and the component cooling water systems will fail. Therefore, the ECI function will fail in addition to the Secondary Cooling function that fails at 19-feet MLW due to flooding of the AFWS. The frequency of core melt in this case is 2.29E-6. This is just the probability that the water level reaches 22-feet MLW.

For the above two cases no credit is taken for operator actions to prevent floodings. This is because the procedures are basically designed to prevent leakage but not a major flood. For example, the emergency procedures do not specifically call for the plant to be shutdown in the event of a hurricane.

6.2 Results

The results of the core melt frequency for the two levels of external flooding at St. Lucie are given in Table 6-1. The largest contribution to core melt is due to flood level of 22-feet MLW.

TABLE 6-1
CORE MELT FREQUENCY CONTRIBUTORS DUE TO EXTERNAL
FLOODING AT ST. LUCIE

Stillwater Flood Level (feet MLW)	Core Melt Frequency
19<x<22	9.21E-7
22<x	2.29E-6

It should be noticed that for the two flood levels, the recovery of offsite power is not a factor. This is because the safety systems themselves are inundated with water. In addition, the diesel generators are available since they are located at 23.5 feet MLW.

It can also be noticed that the frequency of having a flood between 19 feet and 22 feet is just 1.15E-5 (frequency of a flood level that exceeds 19 feet) minus 2.29E-6 (frequency of a flood level that exceeds 22 feet). The frequency of a flood between 19 feet and 22 feet is 9.21E-6. Therefore, the core melt probability due to a flood between 19 feet and 22 feet is 9.21E-6.((frequency of flood level between 19' and 22') times 1E-1 (failure probability of F&B)). This equals 9.21E-7. This would be the amount that the core melt probability would be decreased if the stop logs were reinstated so that the flood level would be increased to 22'.

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0586

APPENDIX G

EXTERNAL WIND ANALYSIS FOR THE ST. LUCIE NUCLEAR POWER PLANT

by

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Jack R. Benjamin & Associates

and

Walter L. Ferrell
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)587

TABLE OF CONTENTS

Page

1.0 INTRODUCTION G-1 589
1.1 Background G-1 589
1.2 External Wind Risk Analysis G-1 589
1.3 Analysis Approach G-3 591
1.4 Report Organization G-3 591

2.0 PLANT AND SITE CHARACTERISTICS G-5 593
2.1 Plant Description and Design Wind Loadings . G-5 593
2.2 Vulnerable Structures and Equipment G-7 595
2.3 Wind Hazard G-8 596

3.0 WIND HAZARD G-14 602
3.1 Methodology G-14 602
3.1.1 Tornadoes G-15 603
3.1.2 Straight Winds G-20 608
3.2 Results G-21 609

4.0 TORNADO MISSILE ANALYSIS G-34 622
4.1 Methodology G-35 623
4.2 Results G-37 625

5.0 SYSTEM ANALYSIS G-40 628
5.1 Methodology G-40 628
5.2 Results G-40 628
5.2.1 Component/System Failures Due to Wind/Tornadoes G-40 628
5.2.2 Wind/Tornado Event Tree G-41 629

6.0 RISK QUANTIFICATION G-45 633
6.1 Methodology G-45 633
6.1.1 Core Melt Probability Contribution Due to Straight Winds G-45 633
6.1.2 Core Melt Probability Contribution Due to Tornado Winds and Tornado Missiles G-46 634
6.1.3 Total Core Melt Probability G-46 634
6.2 Results G-46 634
6.2.1 Straight Wind Results G-47 635
6.2.2 Tornado Winds and Missile Results ... G-47 635
6.3 Conclusions and Results G-49 636

7.0 REFERENCES G-51 639

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1.0 INTRODUCTION

1.1 Background

Sandia National Laboratories (SNL) is participating in the Task Action Plan (TAP) A-45 program to evaluate the adequacy of current licensing requirements for shutdown decay heat removal systems. As part of the SNL study, the response of decay heat removal systems to internal and external initiating events is being considered. The full scope of the analysis plan is reported in Reference 1.

Science Applications International Corporation (SAIC) is assisting SNL in the internal event and external event analyses being conducted for the U.S. Nuclear Regulatory Commission (USNRC). Jack R. Benjamin and Associates, Inc. (JBA) is a subcontractor to SAIC and provided the wind hazard and fragility data used in the TAP A-45 risk analyses. The analyses involving extreme winds for the St. Lucie Plant Unit 1 (referred to as St. Lucie) are given in this report.

1.2 External Wind Risk Analysis

As stated in Reference 1, it is beyond the scope of the TAP A-45 risk program to perform an in-depth probabilistic risk assessment. Instead, the objective is to conduct an analysis that quantifies the significant threats to a plant. However, the analysis procedure used to perform the wind analyses embodies the basic philosophy of a full-scope probabilistic risk assessment.

To evaluate the frequency of plant damage due to external wind events, the following tasks are performed: 1) plant familiarization, 2) tornado and straight wind hazard analyses, 3) tornado missile and wind pressure fragility analyses, 4) systems analysis, and 5) risk quantification. Figure 1-1 shows the relationship of the steps in the wind risk analysis. The systems and risk quantification were performed by SAIC and SNL, while the hazard and fragility analyses were conducted by JBA.

0589

The steps in the analysis are summarized below:

- Plant Familiarization - The purpose of this task is to gather information related to the occurrence of external wind hazards at a plant and the vulnerability of plant structures and equipment to wind effects. In assessing the reliability of shutdown decay heat removal systems, a review of available documentation (i.e., FSAR, plant drawings and topical reports) and a walkdown of the plant are performed.

- Hazard Analysis - Mean wind hazard curves for tornadoes and straight winds were developed by the USNRC for use in the TAP A-45 project. The purpose of the hazard analysis conducted by JBA is to develop the uncertainty in the hazard curves. Independent tornado hazard analyses are conducted which include consideration of the uncertainty in the tornado plant strike model, site size, tornado data, tornado damage area and length variation, and F-scale intensity/windspeed conversion. The resulting curves are scaled so that the resulting mean frequencies of exceedance equal the USNRC values. An analysis of hurricanes and other straight winds is conducted to develop the uncertainty in a similar manner as was done for tornadoes. Uncertainty in the windspeed data bases, applicability of recording station data to the site and terrain roughness differences are considered.

- Tornado Missile and Wind Pressure Fragility Analyses - A tornado missile analysis is conducted by JBA which considers the occurrence frequency of tornadoes, number of available tornado missiles, target exposure, and target area. Results of past tornado missile impact studies which were based on detailed missile simulations are used. For structures and equipment vulnerable to wind pressure hazards, the cumulative fraction of failure defined as a function of windspeed, called a fragility curve, is assessed. Due to uncertainty, multiple fragility curves are determined for each component.

- Systems Analysis - The shutdown decay heat removal systems analysis is performed by developing event trees and fault trees with external wind as the initiating event. The component fragilities and missile impact factors are used to evaluate the conditional frequency of failure of different safety systems.

- Risk Quantification - The quantification of the likelihood of plant damage is assessed by properly weighting the conditional fraction of plant damage (i.e., system fragility curve), defined as a function of wind intensity, by the frequency of occurrence of each windspeed. By propagating the uncertainty in the frequency of wind hazards and the component fragilities, the probability distribution on the frequency of core melt is evaluated.

1.3 Analysis Approach

A conservative approach was used in developing the capacities of the structures and equipment to resist the effects of wind. This procedure is based on the a priori assumption that the individual contribution of each of the components to core melt is not significant. If it is found in the systems analysis that wind hazard is a significant contributor relative to other external and internal initiators, then a more detailed look at components which dominate the wind contribution will be required. By proceeding in this manner the analysis effort is focused on the most significant contributors, and components which are not contributors are screened out by a defensible conservative approach that requires a minimum of effort.

1.4 Report Organization

In Section 2.0, the plant and site characteristics of St. Lucie are described from the perspective of the effects of wind events on shutdown decay heat removal systems. The plant site is described and the critical structure and equipment items vulnerable to wind effects are given.

In Section 3.0, the uncertainty analysis for tornado and straight winds, including hurricanes, is presented. Hazard data and corresponding curves are given.

Section 4.0 presents the results of the tornado missile analysis. Fractions of missile impact for vulnerable components are given.

In Section 5.0 the system analyses are presented. In Section 6.0 the risk quantification is presented.

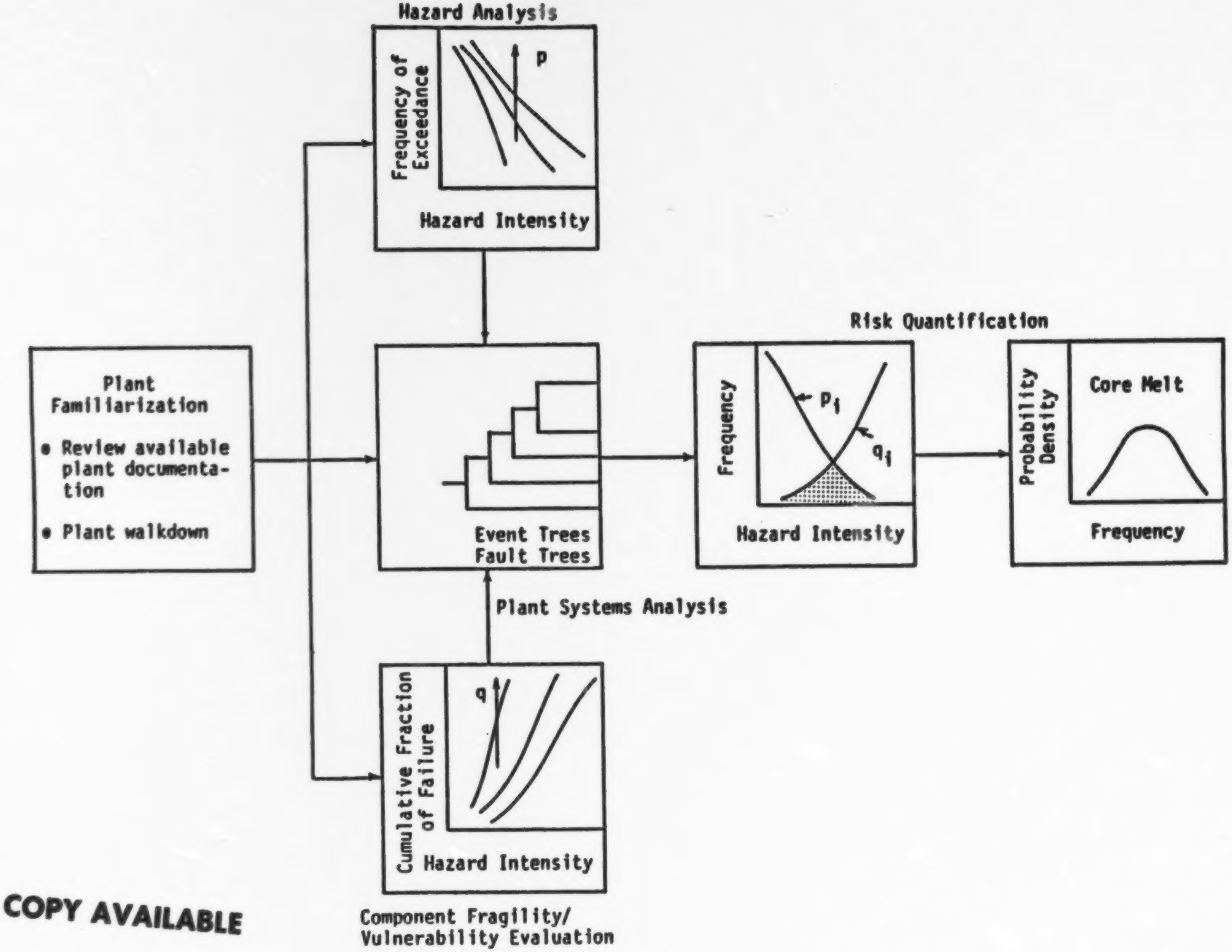
Hazard Analysis
Frequency of Exceedance
p
Hazard Intensity
Plant Familiarization
• Review available plant documentation
• Plant walkdown
Event Trees
Fault Trees
Plant Systems Analysis
Risk Quantification
Frequency
p_i
q_i
Hazard Intensity
Probability Density
Core Melt
Frequency
Cumulative Fraction of Failure
q
Hazard Intensity
Component Fragility/
Vulnerability Evaluation

FIGURE 1-1 RISK ANALYSIS FLOWCHART

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0592

2.0 PLANT AND SITE CHARACTERISTICS

A plant walkdown of St. Lucie was conducted to gain familiarity with the plant and site characteristics and to determine the vulnerable structures and equipment critical to decay heat removal. Copies of criteria and design documents, and construction drawings were obtained and reviewed. Discussions with plant and utility personnel were conducted to obtain an understanding of the structural protection provided to resist the effects of wind.

A description of the plant and site characteristics in regard to the wind analysis is given in Section 2.1. A list of vulnerable structures and equipment which were analyzed are discussed in the following section. Finally, a general description of the wind hazards at the St. Lucie plant site is described in the last section.

2.1 Plant Description and Design Wind Loadings

The structures at St. Lucie are constructed of reinforced concrete or steel framing with metal siding. Figure 2-1 is a plot plan of the site which shows the general arrangement of the structures and location of equipment vulnerable to wind effects. These include the structures and equipment listed in Table 2-1.

The safety-related equipment which is part of the shutdown decay heat removal system is contained within the reactor building, reactor auxiliary building and diesel generator building, which are Class I structures. The condensate storage tank (CST), the refueling water storage tank (RWST) and the diesel fuel oil storage tanks are located at ground level outside of the buildings. Finally, the diesel fuel oil transfer pumps the intake cooling water pumps, and the CCW heat exchanger and pumps are also located outside.

0593

In general, safety-related equipment are protected in Class I structures with exterior concrete walls and roofs at least 12 inches thick. In some locations the walls and roof slabs are thicker. The auxiliary feedwater pumps, intake cooling water pumps and CCW pumps are protected by steel frame structures and 1-inch steel plate or grating.

All Class I structures at St. Lucie were designed as a minimum for a load corresponding to 194 mph windspeed (Ref. 2). In addition, the reactor building, the auxiliary building which encloses the control room, the fuel building, the diesel generator building and all other seismic category I structures needed for the safe shutdown of the plant, are designed for tornado loads stated in References 2 and 3 as follows:

1. A tangential wind velocity of 300 mph.

2. A translational wind velocity of 60 mph (for fuel handling building only).

3. A pressure drop of three psi (except diesel generator building which was designed for 2.25 psi) occurring over a three-second time interval.

In addition, the structures were designed for the following tornado missiles (note that the tornado pressures were not combined with the missile effects):

1. A 2-inch x 4-inch wooden plank 10 feet long at a velocity of 360 mph.

2. A 4000-pound automobile flying through the air at 50 mph.

The capability of 1-inch thick steel, 1/2-inch thick steel and 1-foot thick concrete to resist penetration by several types of tornado missiles is given in Reference 3.

2.2 Vulnerable Structures and Equipment

In general, most safety-related equipment is contained within and protected by concrete structures which are at least 12 inches thick, and in many cases the barriers are thicker or by steel frames and 1-inch thick steel plate or grating. Equipment which is not protected and which is vulnerable to wind effects is listed in Table 2-1. It was assumed in the development of the vulnerable structure and equipment list that concrete walls and roof slabs greater than 12 inches thick and heavy steel plates or grating provide adequate protection against tornado missiles.

Although very large missiles could penetrate 1-inch thick steel plates or grating, it is extremely unlikely that large missiles will impact the safety-related equipment located outside. From a probabilistic viewpoint, the contributions to the likelihood of failure of this equipment by missile impact is not significant.

Doors, blockouts, exhaust stacks and other openings to Class I structures were postulated to be vulnerable entrances to the buildings for tornado missiles. In general, all openings were either found to be protected by barrier walls, tornado missile barriers or too small to be significant contributors to the frequency of core melt. Based on a scoping analysis, openings less than 100 ft^2 correspond conservatively to a mean frequency of tornado missile impact less than 5.0-7[1] per year; thus, the door openings were not found to be significant contributors and were not considered further.

The CST and the diesel oil storage tanks are located outside and are vulnerable to wind effects. These tanks are Class I structures and were not analyzed in this study for wind pressure effects based on experience with similar tanks at other nuclear power plants. Because these tanks are designed for seismic loads, the capacities of the tanks are sufficient to resist wind velocities greater than 360 mph. However the diesel fuel oil storage tanks and the RWST are vulnerable to tornado missiles and were included in the tornado missile analysis.

---

[1] $5.0\text{-}7 = 5.0 \times 10^{-7}$

The intake cooling water pipes and the component cooling water pipes are not vulnerable to wind pressure but they could be damaged by large tornado missiles and were included in the tornado missile analysis.

Based on the design and review criteria (Ref. 3) all the plant Class I structures exposed to wind were designed for a hurricane wind velocity of 194 mph and the safety-related buildings and structures were designed to resist a tornado of 300 mph tangential wind velocity and a 60 mph translational wind velocity (for fuel handling building only). A wind speed of 300 mph corresponds to a mean frequency of occurrence of 1.5-7 per year. Based on this result the capacities of the buildings were not considered further. Thus no detailed wind pressure analysis was required for the St. Lucie plant.

2.3 Wind Hazard

The St. Lucie site is exposed to potential winds generated by hurricanes, tornadoes and extratropical cyclones (including winter storms and summer thunderstorms). Extratropical cyclones and hurricanes are referred to in this report as straight winds.

A listing of all tornadoes within approximately a 145-mile radius of St. Lucie was obtained from the National Severe Storms Forecast Center (NSSFC) in Kansas City, Missouri. From 1950 through 1983, a total of 735 tornadoes were recorded of which none were rated larger than Fujita scale F-3. Two F-4 tornadoes occurred approximately 200

0596

miles north of the St. Lucie site based on a record between 1880 and 1982 (Ref. 4). The locations of the two F-4 events are shown in Figure 2-2.

St. Lucie is located on the east side of Florida on the Atlantic side of the peninsula. The site is located in Region I of the USNRC tornado risk regionalization scheme given in WASH-1300 (Ref. 5). This is the region which has the highest hazard of the three USNRC regions. It is also located in Region B of the 4-region scheme proposed by Twisdale and Dunn as shown in Figure 2-3 (Ref. 6). Region B has a regional occurrence rate corrected for unreported tornadoes (occurrences per square mile per year) of 6.98-4 compared to USNRC Region I which has 4.12-4. However, the percentage of large tornadoes (i.e., F-4 and F-5's) is lower in Region B compared to the larger USNRC Region I.

Straight wind data were obtained from the weather station at the Miami, Florida airport which is approximately 90 miles south from the site (Ref. 7). For the period 1912 to 1961 the largest recorded fastest mile windspeed is reported to be 129 mph, which occurred in 1926 due to a hurricane. Also, hurricane simulation analyses based on Monte Carlo procedures for the Atlantic and Gulf coasts were used (Ref. 8).

TABLE 2-1

VULNERABLE STRUCTURES AND EQUIPMENT

Structure/Equipment	Location*	Comment
CCW Pipes	East of the Fuel Handling building	Exposed to tornado missiles
Diesel Oil Storage Tanks (2)	North East of Diesel Generator building	Vulnerable to tornado missiles
RWST	North of Diesel Generator building	Vulnerable to tornado missiles
ICW Pipes	West of the Turbine Building and East of Fuel Handling building	Exposed to tornado missiles

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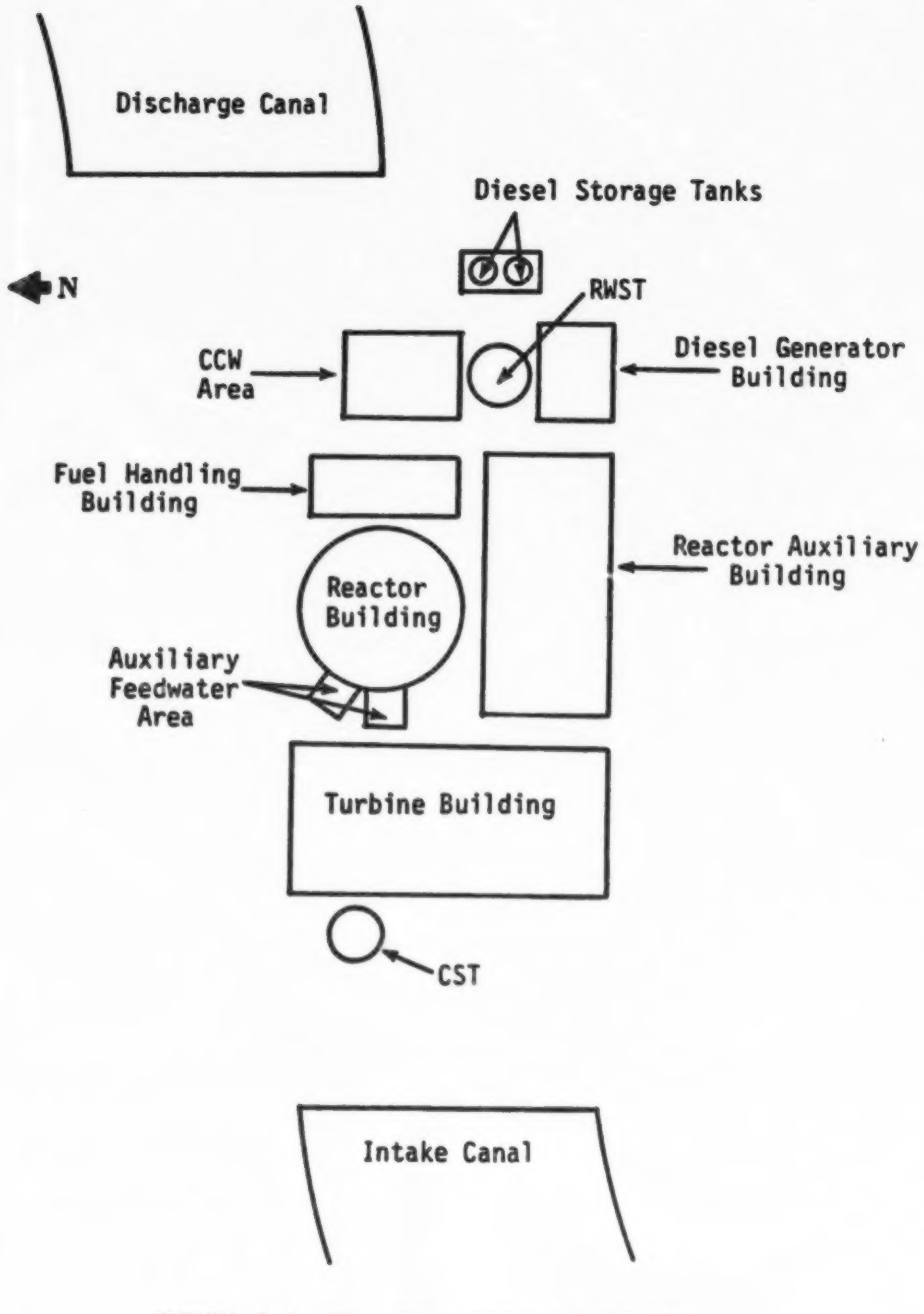
Discharge Canal
Diesel Storage Tanks
N
RWST
CCW Area
Diesel Generator Building
Fuel Handling Building
Reactor Auxiliary Building
Reactor Building
Auxiliary Feedwater Area
Turbine Building
CST
Intake Canal

FIGURE 2-1 ST. LUCIE UNIT 1 PLOT PLAN

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0599

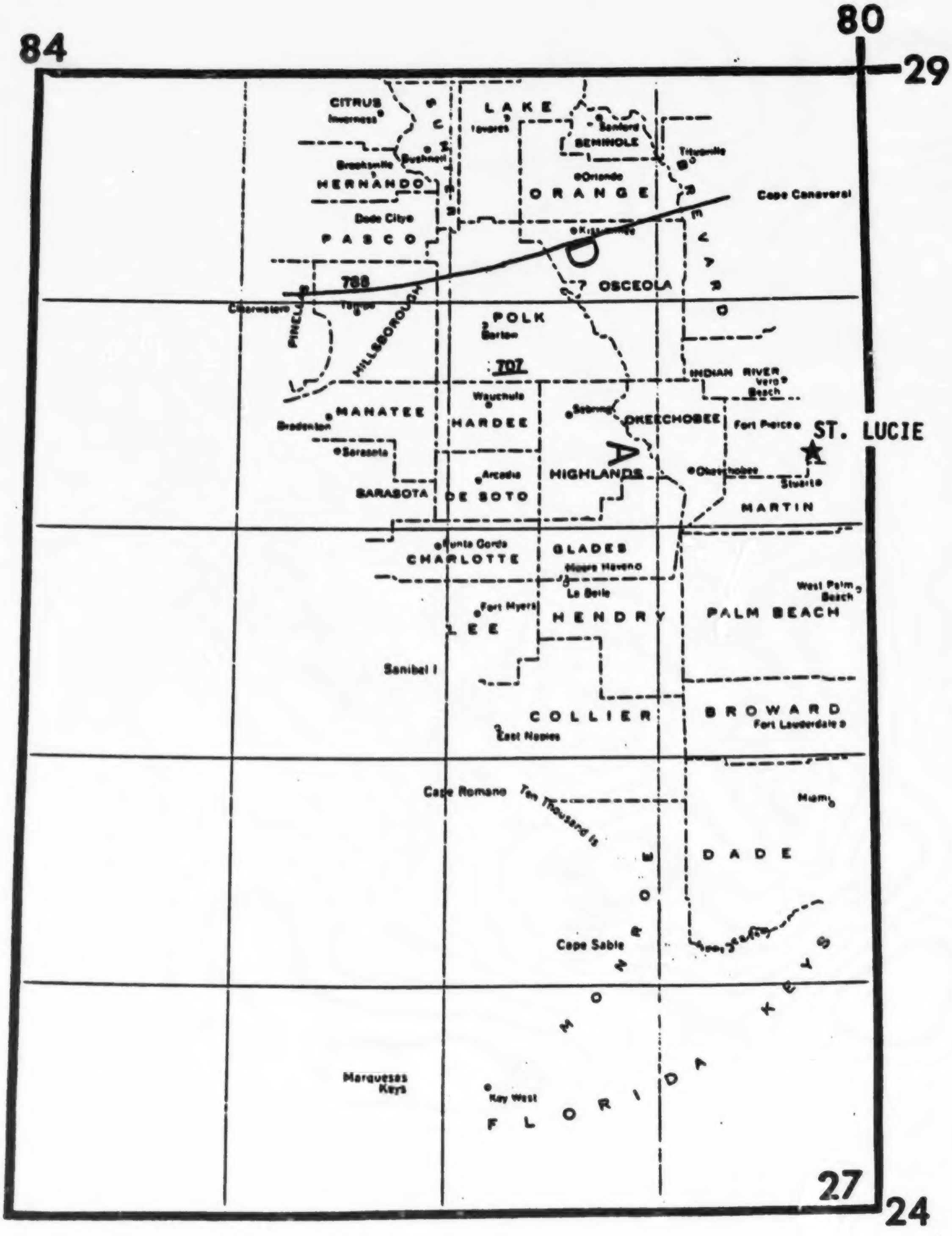
84
80
29
CITRUS
Inverness
LAKE
Tavares
Sanford
SEMINOLE
Titusville
Brooksville
Bushnell
Orlando
HERNANDO
ORANGE
Cape Canaveral
Dade City
PASCO
Kissimmee
D
788
OSCEOLA
Clearwater
Tampa
PINELLAS
HILLSBOROUGH
POLK
Bartow
707
INDIAN RIVER
Vero Beach
Wauchula
MANATEE
Bradenton
HARDEE
Sebring
OKEECHOBEE
Fort Pierce
ST. LUCIE
Sarasota
A
HIGHLANDS
Okeechobee
Stuart
Arcadia
SARASOTA
DE SOTO
MARTIN
Punta Gorda
CHARLOTTE
GLADES
Moore Haven
West Palm Beach
La Belle
Fort Myers
HENDRY
PALM BEACH
LEE
Sanibel I
COLLIER
BROWARD
Fort Lauderdale
East Naples
Cape Romano
Ten Thousand Is
Miami
MONROE
DADE
Cape Sable
FLORIDA KEYS
Marquesas Keys
Key West
FLORIDA
27
24

FIGURE 2-2 HISTORIC F-4 TORNADO TRACES (REF. 4)

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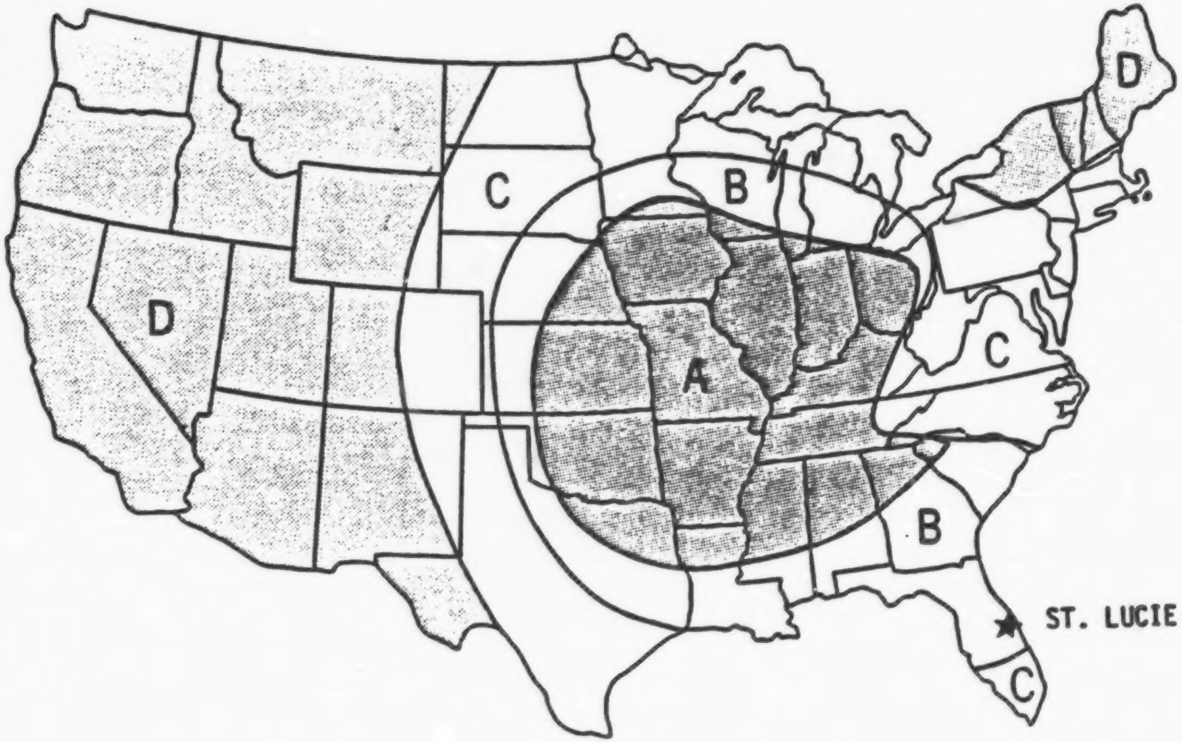
D
C
B
D
A
C
B
ST. LUCIE
C

FIGURE 2-3 TORNADO RISK REGIONALIZATION SCHEME PROPOSED BY TWISDALE AND DUNN (REF. 6)

0601

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3.0 WIND HAZARD

Hazard data were developed for both tornadoes and straight winds (including hurricanes) for the St. Lucie site. Based on an agreement between the USNRC and SNL, the Meteorology Section, Meteorology and Effluent Treatment Branch of the USNRC, prepared best estimate wind hazard curves for the St. Lucie site for both wind types. The basis for this agreement was that the Meteorology Section had an established procedure for determining the frequency of high winds at nuclear power plant sites and that the analyses to be provided by SNL should utilize those frequency assessments to be consistent with other risk analyses requiring wind hazard evaluations.

The objective of the wind hazard analysis performed for St. Lucie by JBA was to develop the uncertainty about the best estimate hazard curves which were assumed to be mean hazard curves (i.e., for tornado and straight winds) and to prepare a family of curves, which have a corresponding mean curve that is equal to the USNRC curve for each of the two wind types.

The next section discusses the methodologies used to calculate the uncertainty in the wind hazard. The last section presents the results of the hazard analysis.

3.1 Methodology

The methodology for the uncertainty analysis performed for tornadoes and straight winds is given in Sections 3.1.1 and 3.1.2, respectively.

3.1.1 Tornadoes

The tornado analysis followed the basic tornado analysis procedure given by Reinhold and Ellingwood (Ref. 10). Uncertainty was considered in the following five parts of the analysis:

- Strike model
- Plant site width
- Tornado analysis data
- Tornado damage area and length variation
- F-scale windspeed conversion

Multiple analyses were performed to develop a family of wind hazard curves. The mean frequency of exceedance was determined at seven windspeeds (i.e., 66, 93, 134, 182, 234, 290, and 349 mph) and compared to the corresponding mean frequency values developed by the USNRC. All hazard curves were shifted by the ratios of calculated mean values to the USNRC values. From the scaled probability distributions at each windspeed, five equal probability frequency of exceedance values were calculated. The mean values of the resulting five frequencies at each windspeed are equal to the USNRC values and have a reasonable distribution which reflects the variabilities developed from a systematic consideration of the underlying uncertainties in the models and parameters.

Figure 3-1 shows the uncertainty analysis flowchart. The probability weights subjectively assigned to each branch are shown. The basis for the five parts of the analysis is discussed below.

Tornado Plant Strike Model

Two strike models were considered. Figure 3-2 shows the geometric relationships for the point strike and union strike models. The point strike model considers the site to be represented by a point and requires that any part of the tornado which crosses the site leads to a strike. Thus the tornado strike area is equal to the area enclosed by the dashed centerline shown in Figure 3-2 which is simply equal to the tornado width, W, times the tornado length, L.

In contrast, the union strike model considers the site to be represented by a rectangular area of width, A, and length, B. Again if any part of the tornado crosses any part of the site, a strike occurs. The tornado strike area is equal to the area enclosed by the dashed centerline as shown in Figure 3-2, which is approximately equal to WL plus AL. Note that the areas equal to BW plus AB have been neglected since they are small relative to the area retained. By formulating the model in this manner the union strike model is equal to the sum of the point strike model and a "lifeline" term which is proportional to the width of the site.

Plant Site Width

The second part of the analysis shown in Figure 3-1 which was considered to be uncertain is the plant site width (see dimension A in Figure 3-2). Since the direction of a tornado strike is not known the width varies unless the site is a perfect circle. Three site widths were considered (200, 350 and 500 feet). The lower value simulates the possibility that the tornado must overlap the site for damage to occur, while the upper value conservatively includes all possible structures at the site. The middle value represents a diagonal across the area which contains the major safety-related structures.

Tornado Analysis Data

The tornado analysis data considered both local and regional data. As indicated in Table 3-1 the tornado analysis data consist of the occurrence fraction of F-scale tornado intensities, the average tornado area and average tornado length corresponding to each intensity, and

0604

finally the rate of tornado occurrence in the area surrounding the site. The regional data corresponds to Region B, which contains the St. Lucie site, as shown in Figure 2-3. These data are given in Reference 10.

The local data were based on the tornadoes recorded in the NSSFC data base from 1952 through 1983 within a radius of 145 miles of the site (see discussion in Section 2). The tornado rate of occurrence and distribution of F-scale intensities were obtained from these data, but considered reporting trends, direct classification and random encounter errors, and unrated tornadoes (see Reference 10 for a discussion of these issues). The average tornado damage area and path length for the local area are based on the NSSFC data for intensities less than F-4. Since no F-4 and F-5 events have occurred near the site, the regional data were used. Note that no F-6 tornadoes have ever been recorded, hence the F-6 data are based on extrapolation.

Tornado Damage Area and Length Variation

A tornado rated by the Fujita F-scale is based on the highest level of damage which occurs within the total tornado damage area. Figure 3-3 shows a sketch of a hypothetical F-4 tornado illustrating the variation of intensity within the total damage area. Note that only a small portion of an F-4 rated tornado is actually F-4 damage. There are various relationships which divide the total damage area into the individual lower F-scale damage areas. Table 3-2 shows three possible relationships. For each matrix shown the rows correspond to the actual damage state while the columns refer to the reported damage state. For example in the first matrix (Rheinhold 1) of Table 3-2, 62.5 percent of the area of an F-6 rated tornado is actually F-0 damage.

The Reinhold 1 matrix is based on a variation of intensity along the tornado length using the percentage of damaged length for each tornado analyzed along with an empirically derived damage width relationship. This matrix is considered to be the most realistic. The

0605

Reinhold 2 matrix is based on a variation of intensity along the tornado length based on dividing the tornadoes into 2 to 3 mile segments and assigning the highest intensity observed in each segment as the intensity of the whole segment. The damage width relationship for the second matrix is based on a theoretical wind model with a tornado translation speed of 59 mph.

The third relationship considers the entire damage to be equal to the maximum damage (very conservative). This approach was used in WASH-1300 (Ref. 5) and is assumed in the USNRC procedure for determining wind hazard.

Table 3-3 gives the corresponding matrices for the damage length relationships used for the lifeline term calculation. The corresponding assumptions as discussed for the area relationships were used in the development of the matrices given in Table 3-3.

F-Scale Windspeed Conversion

The last part which was varied in the uncertainty analysis is the conversion between the F-scale intensities and the maximum windspeed. There have been discussions in recent years that the windspeeds originally suggested by Fujita are too high (Refs. 9 and 10). In addition to the Fujita conversion, two additional conversions were assumed as given in Table 3-4. These were developed by Twisdale and are referred to as the "linear" and "uniform" cases. In developing these two cases, the Fujita windspeeds were used as a prior and were updated from data which give upper-limit values of the windspeed ranges. Failure calculations were performed and limiting windspeed bounds were calculated. For the uniform case the windspeed range was assumed to be uniform, while for the linear case, it was assumed to vary linearly from the upper bound to a zero likelihood at the lower bound.

0606

Tornado Windspeed Frequency Analysis

The equations for a tornado hazard analysis, based on a single set of model assumptions, for the point strike and lifeline terms are given by the following equations:

Point Strike: $F(V) = OR[CVLVW][AI \cdot OI]$ (3.1)

Lifeline: $F(V) = OR[VL][LI \cdot OI]A$ (3.2)

where: $F(V_i)$ = Frequency of occurrence of tornadoes with a maximum windspeed equal to V_i. $F(V)$ is the vector of values for the seven windspeeds.

OR = Occurrence rate of tornadoes (see Table 3-1)

[CVLVW] = Combined area variation matrix (see Table 3-2)

[AI] = Average tornado area vector (see Table 3-1)

[OI] = F-scale occurrence fraction vector (see Table 3-1)

[VL] = Damage length variation matrix (see Table 3-3)

[LI] = Average tornado length vector (see Table 3-1)

A = Width of plant site (see Figure 3-2b)

The wind hazard curve is just the frequency of exceeding a windspeed V_j as given by the following equation:

$$F(V_j) = \sum_{i=j}^{n} F(V_i) \qquad (3.3)$$

0607

Figure 3-4 shows an example analysis in expanded matrix form. The example shown corresponds to the top branches in the analysis flowchart. The last column contains the frequencies of occurrence corresponding to the windspeeds given in the first column. By successively accumulating the frequencies from the bottom of the last column, the hazard curve values are obtained (i.e., as given by Equation 3.3).

3.1.2 Straight Winds

The straight wind analysis combined the occurrences of hurricanes and extratropical cyclones (i.e., normal storms and thunderstorms). The approach used considered the available data, which included historic data recorded at the Miami weather station (Ref. 7), results of hurricane simulation studies along the Atlantic and Gulf coasts (Ref. 8), and a previous hurricane hazard study conducted for the Indian Point Nuclear Power Plant (Ref. 19). The three-step procedure for combining these data is described below.

First the data from the Miami weather station were separated into tropical and extratropical groups. Forty percent of the maximum annual windspeeds are tropical and sixty percent are extratropical. The extratropical data were fit to an extreme value Type I distribution (Ref. 11) while the tropical data were fit to a Weibull distribution (Ref. 11) as recommended by Changery (Ref. 7). In parallel, a mean hazard curve based on Monte Carlo simulations of hurricane storm tracks was obtained from Reference 8. Since this reference does not provide an uncertainty estimate, the results of a recent hurricane study conducted at Indian Point were used to obtain the coefficients of variation (COV) on the windspeed at difference frequencies of exceedance (Ref. 19). Combining the COV values with the mean hazard curve produced an independent assessment of the hurricane hazard at St. Lucie. This assumed that the uncertainty at Indian Point and St. Lucie is the same.

In the second step, the two hurricane hazard estimates (i.e., based on weather station data and simulations) were probabilistically combined based on equal weight. In this step the distribution of windspeeds at each frequency of exceedance were considered to be normally distributed.

Finally in the last step the extratropical data which were fit to the Type I distribution were probabilistically combined with the resulting hurricane hazard curves from step 2. For this combination the extratropical and tropical hazard curves were weighted 0.60 and 0.40, respectively. The final hazard curves were shifted so that the mean curve was the same as the curve developed by the USNRC, but the COV obtained from the analysis described above was preserved.

3.2 Results

As indicated by Figure 3-1 a total of 72 hazard curves were run for the tornado analysis with corresponding probability values which varied from 0.0002 to 0.12. The probability distribution on the frequency of exceedance was shifted at each frequency so that the mean frequency values were equal to those provided by the USNRC. By synthesis, 5 hazard curves, each with equal probability values (i.e., 0.20) were calculated. Table 3-5 gives the hazard data including the mean frequency values and Figure 3-5 shows a plot of the 5 hazard curves. Note that the last row in Table 3-5 gives the frequency of occurrence values for any size tornado strike. These values are needed for the tornado missile impact analysis.

Five equal probability hazard curves were calculated for the straight winds based on the analysis procedure described in Section 3.1.2. The mean values are the same as the values calculated by the USNRC. Table 3-6 gives the hazard data including the mean frequency values, and Figure 3-6 shows a plot of the 5 hazard curves for straight winds.

0609

TABLE 3-1

TORNADO ANALYSIS DATA

	Regional Data (Ref. 9)			Local Data		
F-Scale	Occurrence Fraction [OI]	Average Area (mi^2) [AI]	Average Length (mi) [LI]	Occurrence Fraction [OI]	Average Area (mi^2) [AI]	Average Length (mi) [LI]
F-0	.2263	0.30	1.53	.4010	0.02	0.42
F-1	.3527	0.72	3.01	.3774	0.06	1.81
F-2	.2785	1.80	4.76	.1797	0.18	3.98
F-3	.1040	4.30	9.15	.0376	1.17	7.12
F-4	.0312	8.50	18.80	.0033	8.00	18.80
F-5	.0063	15.70	26.90	.0005	15.70	26.90
F-6	.0008	18.90	30.10	4.4-5	18.90	30.10

Rate of Occurrence, OR: 6.98-4/mi^2/yr   1.20-3/mi^2/yr

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TABLE 3-2

COMBINED AREA VARIATION MATRIX, [CVCVW]

Reinhold 1 (Ref. 10)

Local Tornado State	True Maximum Tornado State						
	F-0	F-1	F-2	F-3	F-4	F-5	F-6
F-0	1.000	.743	.658	.615	.637	.632	.625
F-1	0	.257	.248	.267	.234	.236	.238
F-2	0	0	.094	.091	.093	.088	.089
F-3	0	0	0	.027	.028	.033	.033
F-4	0	0	0	0	.008	.009	.011
F-5	0	0	0	0	0	.002	.003
F-6	0	0	0	0	0	0	.001

Reinhold 2 (Ref. 10)

Local Tornado State	True Maximum Tornado State						
	F-0	F-1	F-2	F-3	F-4	F-5	F-6
F-0	1.000	.751	.484	.316	.315	.293	.289
F-1	0	.249	.414	.423	.371	.330	.326
F-2	0	0	.102	.205	.199	.201	.176
F-3	0	0	0	.056	.089	.112	.107
F-4	0	0	0	0	.026	.050	.063
F-5	0	0	0	0	0	.014	.030
F-6	0	0	0	0	0	0	.009

WASH-1300 (Ref. 5)

1.0 along diagonal; zero elsewhere

TABLE 3-3

DAMAGE LENGTH VARIATION MATRIX, [VL]

Reinhold 1 (Ref. 10)

Local Tornado State	Recorded Tornado State						
	F-0	F-1	F-2	F-3	F-4	F-5	F-6
F-0	1.000	.383	.180	.077	.130	.118	.100
F-1	0	.617	.279	.245	.131	.125	.110
F-2	0	0	.541	.310	.248	.162	.120
F-3	0	0	0	.368	.234	.236	.160
F-4	0	0	0	0	.257	.187	.200
F-5	0	0	0	0	0	.172	.150
F-6	0	0	0	0	0	0	.160

Reinhold 2 (Ref. 10)

Local Tornado State	Recorded Tornado State						
	F-0	F-1	F-2	F-3	F-4	F-5	F-6
F-0	1.000	.572	.280	.116	.142	.133	.120
F-1	0	.428	.352	.245	.158	.102	.130
F-2	0	0	.368	.318	.278	.189	.150
F-3	0	0	0	.321	.210	.242	.165
F-4	0	0	0	0	.212	.185	.175
F-5	0	0	0	0	0	.149	.140
F-6	0	0	0	0	0	0	.120

WASH-1300 (Ref. 5)

1.0 along diagonal; zero elsewhere

TABLE 3-4

F-SCALE WINDSPEED CONVERSION

F-Scale	Windspeed (mph)		
		Twisdale (Ref. 9)	
	Fujita	Linear	Uniform
F-0	40-72	40-73	40-65
F-1	73-112	73-103	65-96
F-2	113-157	103-135	96-114
F-3	158-206	135-168	134-139
F-4	207-260	168-209	139-181
F-5	261-318	209-277	181-236
F-6	319-380	295*	289*

* Mean value extrapolated

TABLE 3-5

TORNADO HAZARD DATA

Windspeed (mph)	Mean Value*	Frequency of Exceedance Per Year				
		Probability Value				
		0.20	0.20	0.20	0.20	0.20
66	1.70-4	5.55-5	6.55-5	1.18-4	1.41-4	4.70-4
93	9.32-5	2.12-5	3.64-5	6.64-5	8.98-5	2.52-4
134	3.31-5	3.42-6	8.03-6	1.75-5	3.25-5	1.04-4
182	6.65-6	1.80-7	7.35-7	1.87-6	4.15-6	2.63-5
234	1.24-6	8.09-9	5.03-8	1.66-7	3.69-7	5.58-6
290	2.03-7	2.57-10	2.28-9	1.14-8	2.86-8	9.72-7
349	2.50-8	8.72-12	1.11-10	6.80-10	2.17-9	1.22-7
Any Size Tornado	1.70-4	5.55-5	6.55-5	1.18-4	1.41-4	4.70-4

* Equal to the USNRC mean values for St. Lucie

0614

TABLE 3-6

STRAIGHT WIND HAZARD DATA

Windspeed (mph)	Mean Value*	Frequency of Exceedance Per Year Probability Value				
		0.20	0.20	0.20	0.20	0.20
80	2.18-1	2.31-2	7.29-2	1.00-1	1.25-1	7.69-1
90	1.01-1	8.73-3	3.81-2	5.96-2	7.86-2	3.20-1
111	2.00-2	1.11-3	6.84-3	1.53-2	2.29-2	5.39-2
120	1.00-2	2.66-4	3.45-3	7.02-3	1.37-2	2.56-2
128	5.00-3	5.32-5	1.37-3	3.64-3	7.73-3	1.22-2
138	2.00-3	1.02-5	4.54-4	1.52-3	3.05-3	4.97-3
144	1.00-3	3.47-6	1.28-4	7.38-4	1.36-3	2.77-3
150	5.52-4	6.45-7	5.36-5	3.78-4	7.45-4	1.58-3
160	2.65-4	1.10-7	9.90-6	1.49-4	3.54-4	8.12-4
174	1.00-4	1.67-8	1.77-6	3.37-5	1.31-4	3.34-4
197	2.00-5	3.03-11	6.11-8	3.18-6	2.40-5	7.28-5
207	1.00-5	1.50-12	1.44-8	1.25-6	1.14-5	3.74-5
230	2.00-6	1.47-15	4.59-10	1.40-7	2.01-6	7.85-6
240	1.00-6	7.54-17	4.10-11	5.57-8	9.48-7	4.00-6

* Equal to USNRC mean values for St. Lucie

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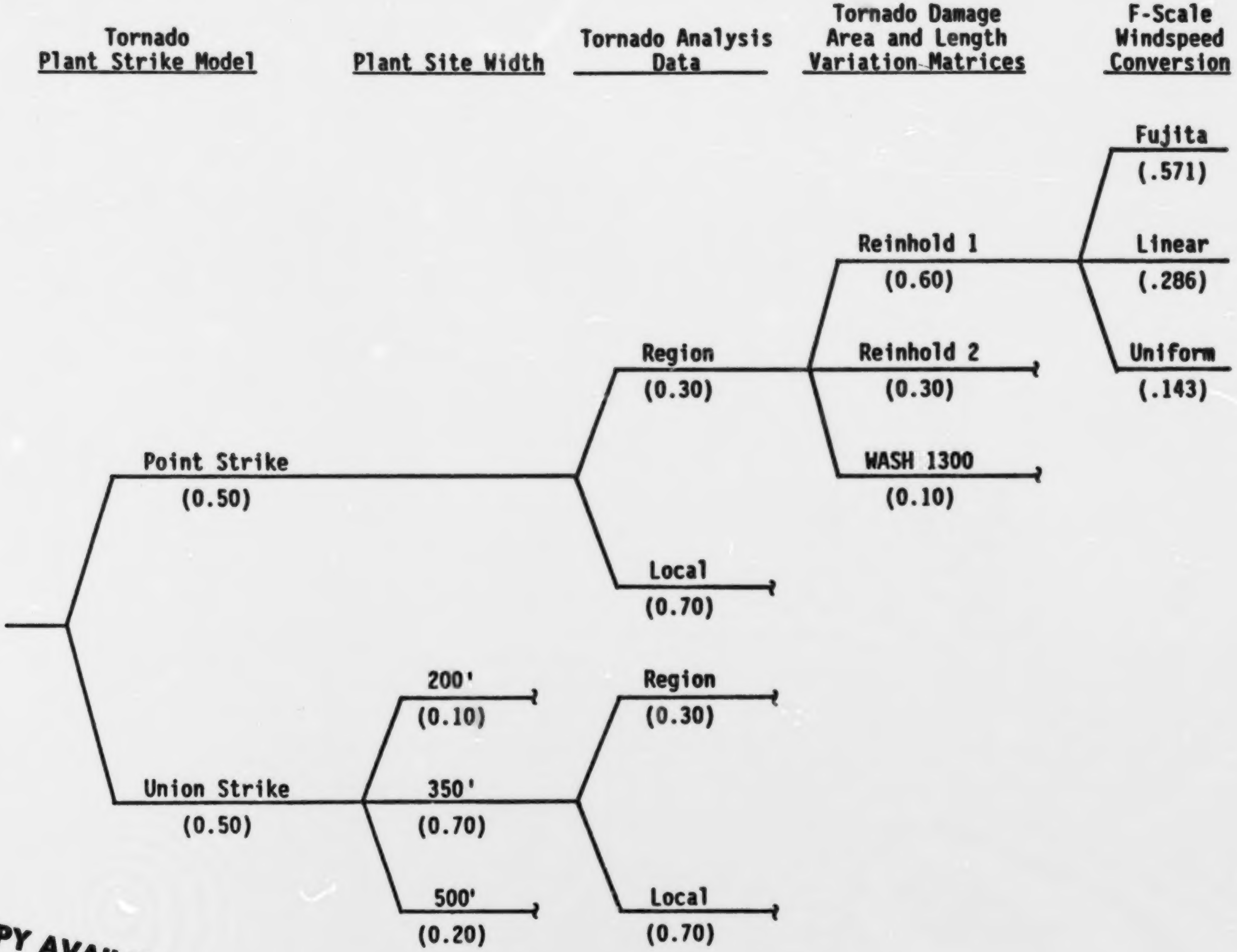

Tornado Plant Strike Model
Plant Site Width
Tornado Analysis Data
Tornado Damage Area and Length Variation Matrices
F-Scale Windspeed Conversion
Fujita
(.571)
Reinhold 1
(0.60)
Linear
(.286)
Region
(0.30)
Reinhold 2
(0.30)
Uniform
(.143)
Point Strike
(0.50)
WASH 1300
(0.10)
Local
(0.70)
200'
(0.10)
Region
(0.30)
Union Strike
(0.50)
350'
(0.70)
500'
(0.20)
Local
(0.70)

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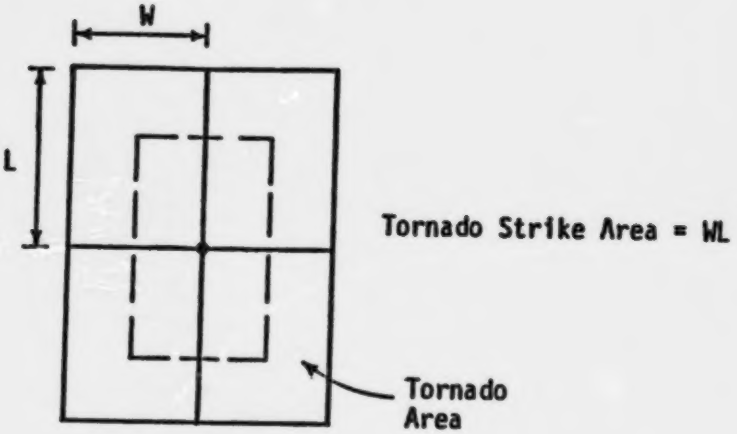
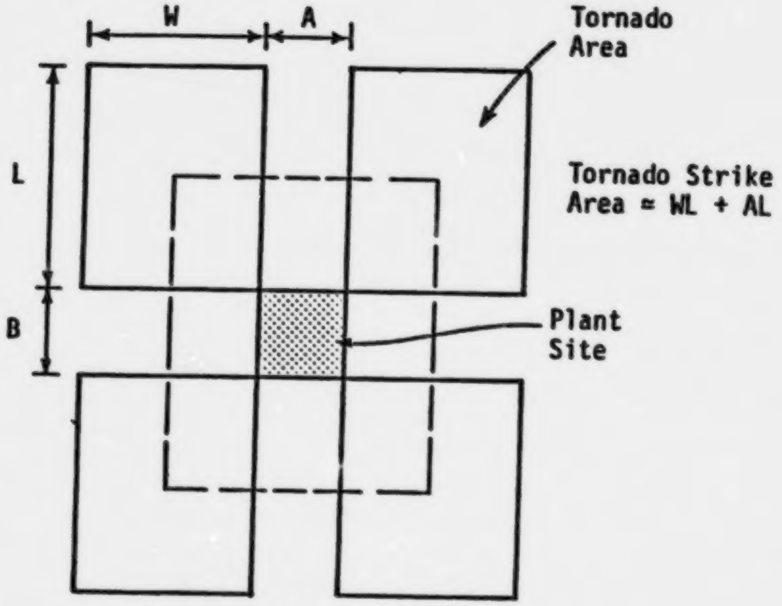
FIGURE 3-1 UNCERTAINTY ANALYSIS FLOWCHART


W
L
Tornado Strike Area = WL
Tornado Area

(a) Point Strike


W
A
Tornado Area
L
Tornado Strike Area ≃ WL + AL
B
Plant Site

(b) Union Strike

FIGURE 3-2 TORNADO STRIKE MODELS

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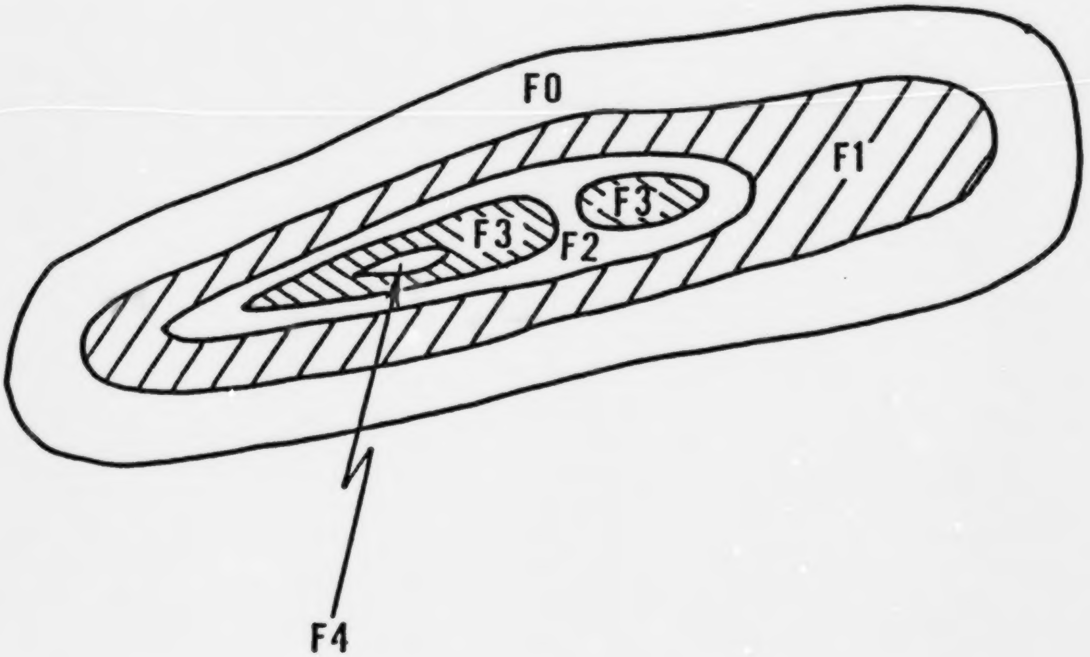
F0
F1
F3
F3
F2
F4

FIGURE 3-3 SKETCH OF HYPOTHETICAL F-4 TORNADO ILLUSTRATING VARIATION OF INTENSITY (REF. 10)

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0618

$$F\begin{bmatrix}56\text{ mph}\\93\\134\\182\\234\\290\\349\end{bmatrix} = F\begin{bmatrix}F\text{-}0\\F\text{-}1\\F\text{-}2\\F\text{-}3\\F\text{-}4\\F\text{-}5\\F\text{-}6\end{bmatrix} \underset{\text{OR}}{= \begin{matrix}6.980\text{-}4\\ \text{Per Year}\end{matrix}} \; \underset{\text{CVLVW}}{\begin{bmatrix}1.000 & .743 & .658 & .615 & .637 & .632 & .625\\ 0 & .257 & .248 & .267 & .234 & .236 & .238\\ 0 & 0 & .094 & .091 & .093 & .088 & .089\\ 0 & 0 & 0 & .027 & .028 & .033 & .033\\ 0 & 0 & 0 & 0 & .008 & .009 & .011\\ 0 & 0 & 0 & 0 & 0 & .002 & .003\\ 0 & 0 & 0 & 0 & 0 & 0 & .001\end{bmatrix}} \underset{\text{AI} \cdot \text{OI}}{\begin{bmatrix}.30 & & .2263\\ .72 & & .3527\\ 1.8 & & .2785\\ 4.3 & X & .1040\\ 8.5 & & .0312\\ 15.7 & & .0063\\ 18.9 & & .0008\end{bmatrix}} = \underset{\text{F(V)}}{\begin{bmatrix}1.13\text{-}3\\2.78\text{-}4\\8.57\text{-}5\\1.62\text{-}5\\2.22\text{-}6\\1.70\text{-}7\\1.06\text{-}8\end{bmatrix}} \begin{matrix}\text{Per}\\ \text{Year}\end{matrix}$$

FIGURE 3-4 EXAMPLE ANALYSIS FOR TOP BRANCHES ON FIGURE 3-1

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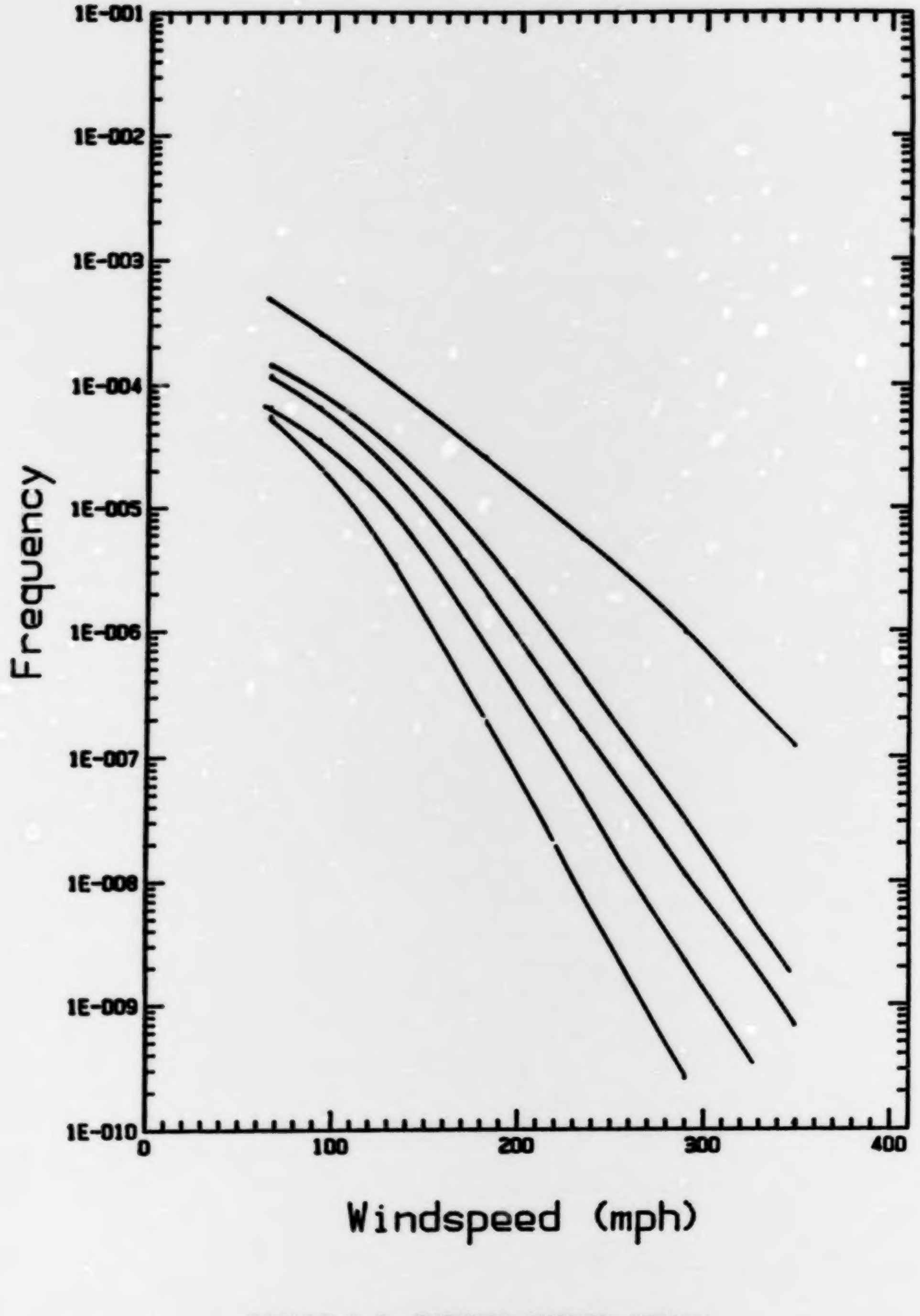

FIGURE 3-5 TORNADO HAZARD CURVES

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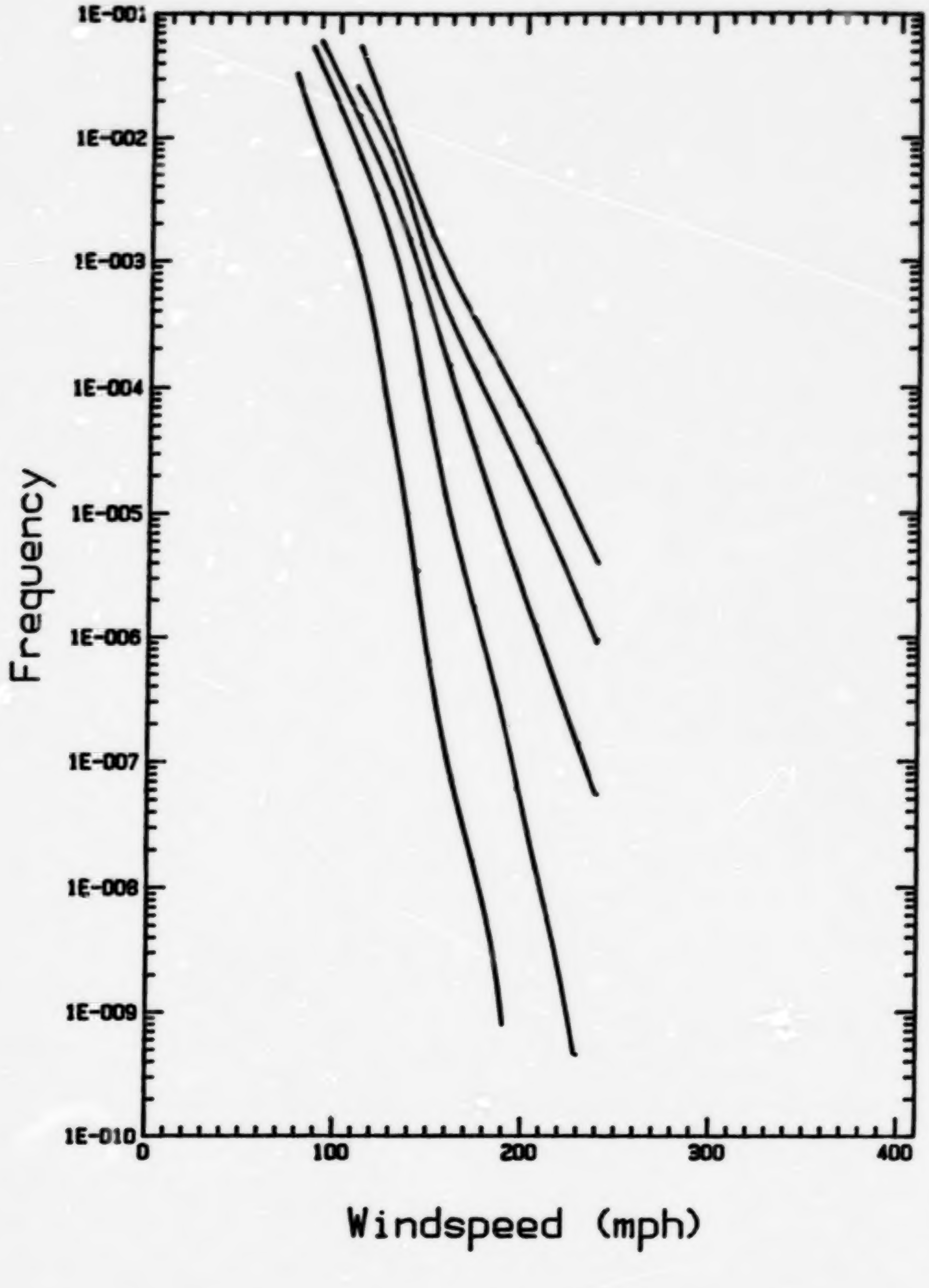

FIGURE 3-6 STRAIGHT WIND HAZARD CURVES

0621

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4.0 TORNADO MISSILE ANALYSIS

A tornado missile analysis was performed to develop the probability distribution on the conditional fraction of missile impact on exposed equipment, which could fail due to missile penetration.

The process of missile impact starts with the injection of a missile into the wind stream which is carried through the air where it could impact exposed equipment or a structure which contains safety-related equipment. The damage caused by a missile is a function of its mass, terminal velocity, deformation characteristics, and angle of impact. The last variable is extremely important since a head-on impact is more damaging than a glancing or side blow.

For structures which contain equipment, several additional events are required to complete the impact scenario. The missile must either penetrate the structure and travel further and impact the equipment, or in the case of concrete, the backside of the wall must scab and a piece of concrete fly through the air and strike the equipment. At each step the probability that a missile (or scabbed piece of concrete) impacts the equipment is less than one, and in simple terms the failure fraction is the product of individual fractions of occurrence corresponding to the events described above.

The methodology used to perform the tornado missile analysis is given in Section 4.1 and the results of the analysis for St. Lucie is presented in Section 4.2.

4.1 Methodology

A simplified tornado missile analysis procedure was developed similar to the analysis conducted for the Oconee PRA (Ref. 12). A normalized tornado missile impact parameter, Ψ, was defined as follows:

Ψ = Frequency of impact/missile/unit target area/tornado point strike frequency

From past tornado missile impact studies, values of Ψ were calculated. It was postulated that the normalized tornado missile impact parameter depends on the following variables:

- Size of target
- Relative F-scale distribution in the region surrounding the site
- Type and location of missiles
- Arrangement of buildings and location of missile targets

Tornado simulation studies using the TORMIS computer code developed by EPRI (Refs. 6 and 13) were used as the basis for calibrating the simplified analysis. Reference 13 gives missile impact values for two plants (A and B1) which were used to develop Ψ-values for large targets (i.e., buildings, tanks, etc.). When normalizing the data for the two plants by USNRC Regions I, II, and III, it was found that the results for the three regions were within a factor of 1.5 with even a smaller difference between Regions I and II. This indicates that the relative F-scale distribution between regions does not have a significant effect on the -value. Parameter values were calculated for all the buildings at the two plants using the slightly higher Region I data and combined into a common distribution; thus, uncertainty in the type and location of missiles and arrangement of buildings was reflected in the variability of the final results.

Table 4-1a gives the resulting normalized tornado missile impact parameter values for large structures in USNRC Regions I and II for 5 equal probabilities. The values range over an order of magnitude with a mean value equal to 1.23-10/missile/ft^2 of target area/tornado point strike frequency. The values of Ψ are normalized with respect to a point strike frequency since this is the basis that the USNRC hazard analysis used. The values in Table 4-1a are applicable for large structures and tanks.

A tornado missile analysis conducted for the Seabrook Station developed impact values for small targets approximately 100 to 1000 ft^2 in size (Ref. 14). Normalized tornado missile impact parameters were developed for each of the targets and the values were found to range from 7.44-13 to 1.17-8/missile/ft^2 of target area/tornado point strike frequency. This range covers over four orders of magnitude. However, the analysis showed that the exposure of the target relative to missile location has a significant effect on the normalized tornado missile impact parameter. In essence, the large structure values given in Table 4-1a represent average values from small areas with some Ψ-values being large and others small. Thus the variability for large structures was not found to be as large as for small target areas.

The 30 target areas in the Seabrook study were segregated into high, medium and low exposure areas. The process of grouping the targets was based on judgment and considered where the targets were located relative to the concentration of missiles and shielding effects of adjacent buildings. Table 4-1b summarizes the results that were found. The variability of Ψ-values for each of the three categories is reduced from the case where all data are considered in one group and is comparable with the variability found for the large structure data. In addition, the large structure values were found to be slightly higher than the medium exposure values for small target areas, which is consistent.

The Ψ-values in Table 4-1 were used to determine the likelihood of missile impact conditional on tornado occurrence. The equipment target areas were conservatively determined and considered to be known with certainty and are given in Table 4-2. An allowance for piping connected to tanks was also included in the tank areas. For each of the targets a conservative distribution on exposure was assumed to be 0.1, 0.4 and 0.5 for large, medium and small exposures. Also, the number of tornado missiles was conservatively estimated to be 5,000, 25,000, and 60,000 with corresponding probability weights of 0.2, 0.6 and 0.2. This distribution of missile inventory is based on judgment and comparisons of the general missile environment at St. Lucie to more accurate counts of missiles at other nuclear power plants. All components except the RWST, and the diesel fuel oil storage tanks were analyzed using the small target area data, while the conditional fractions for the larger equipment are based on the large target area data in Table 4-1.

4.2 Results

Table 4-2 gives the fraction of failure for missile impact conditional on the occurrence of a tornado. Five equal probability values (i.e., 0.20) and the mean value are given for each component. The frequency of occurrence of tornadoes is given in the last row of Table 3-5.

The failure fractions in Table 4-2 can be considered to be independent. For systems in parallel, detailed tornado missile impact studies show that for two components in parallel the conditional fractions can be multiplied together if the reference tornado frequency of occurrence corresponds to a point strike frequency. The tornado occurrence frequency values given in the last row of Table 3-5 are based on a point strike frequency analysis; thus, the missile impact values given in Table 4-2 can be considered to be independent when used for parallel components in a systems analysis.

0625

TABLE 4-1

NORMALIZED TORNADO MISSILE IMPACT PARAMETER VALUES (Ψ)

a) Large Structures in USNRC Regions I and II

Value	Probability	Normalized Parameter, Ψ*
1	0.20	1.49-11
2	0.20	3.49-11
3	0.20	7.16-11
4	0.20	1.61-10
5	0.20	3.31-10
Mean		1.23-10

b) Small Target Areas in USNRC Regions I and II

Value	Probability	Normalized Parameter, Ψ* by Exposure		
		High	Medium	Low
1	0.20	1.42-10	1.30-11	1.61-12
2	0.20	4.03-10	2.52-11	4.09-12
3	0.20	1.63-9	4.86-11	7.71-12
4	0.20	2.57-9	1.14-10	1.46-11
5	0.20	7.33-9	2.31-10	4.87-11
Mean		2.42-9	8.64-11	1.54-11

* Units are per missile per ft^2 of target area per tornado point strike frequency

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0626

TABLE 4-2

CONDITIONAL FRACTION OF FAILURE FOR MISSILE IMPACT

	Probability Weight						
Component	0.20	0.20	0.20	0.20	0.20	Mean Value	Target Area (ft^2)
CCW Pipes	6.93-5	2.72-4	7.31-4	2.21-3	5.08-2	1.07-2	1500
ICW Pipes	6.93-5	2.72-4	7.31-4	2.21-3	5.08-2	1.07-2	1500
RWST	1.03-3	2.94-3	6.18-3	1.45-2	4.46-2	1.38-2	4500*
Diesel Fuel Oil Storage Tank (each)	1.60-4	4.57-4	9.61-4	2.25-3	6.93-3	2.15-3	700**

* Protected on one side by diesel generator building, and by 2 feet high wall. Roof is not included.

** Tanks protected to 2/3 of height on one side by the fuel transfer pump enclosures. Roof is not included.

5.0 SYSTEM ANALYSIS

Tornado missiles, tornado winds, and straight winds can cause failure of various systems in the plant. In this section the response of the plant to the various system failures is modeled. This model is used in Section 6.0 for quantifying the risk due to extreme winds.

The methodology used to perform the system analysis is given in Section 5.1. The results of the analysis for St. Lucie are given in Section 5.2.

5.1 Methodology

To determine the contribution (if any) that tornado winds and missiles, and straight winds can have to core melt probability, the following steps are taken:

- System failures that will occur as a result of vulnerable equipment/structure failures are determined.
- An event tree is constructed that models the plant's response to potential system failures assuming a simultaneous loss of offsite power (LOSP). The assumption that offsite power fails simultaneously is considered realistic for the severity of the storms being analyzed. The LOSP event tree in the internal system analyses is used as a model in constructing the wind/tornado event tree.

5.2 Results

The results for the wind/tornado systems analysis are given in the following two subsections.

5.2.1 Component/System Failures Due to Winds/Tornadoes

The systems that will fail as a result of the failure of the various vulnerable equipment/structures are shown in Table 5-1. The assumption is made that a simultaneous loss of offsite power (LOSP) occurs during a wind/tornado initiated transient event. The vulnerable equipment structures listed in the left column are obtained from Table 2-1. Each system(s) failure is discussed below:

(1) HPIS/LPIS/LPRS Failure

(a) CCW Pipes - Failure of the Component Cooling Water (CCW) pipes will cause failure of the CCW system. This will result in the loss of cooling to all of the pumps for the Emergency Coolant Injection (ECI) systems.

(b) Diesel Oil Storage Tanks - Following an LOSP, failure of the diesel generators will result in the loss of all emergency onsite AC power. This will fail both early (HPIS/LPIS) and late (HPRS/LPRS) ECI systems. The AFWS trains can either fail randomly or due to battery or CST depletion.

(c) ICW Pipes - Failure of the ICW pipes will lead to the failure of the ICW system. This will result in the ECI system failures. This is because the ICW system either directly or indirectly, by cooling the CCW system heat exchangers, provides cooling to all the pumps for the ECI systems.

(2) HPIS/LPIS/CSIS - The RWST is the water source for HPIS, LPIS and CSIS; therefore failure of the RWST will cause failure of the early ECI systems as well as the containment spray injection system.

5.2.2 Wind/Tornado Event Tree

A functional event tree that models the response of St. Lucie to winds and/or tornadoes is shown in Figure 5-1. A simultaneous LOSP is assumed to occur. In Boolean equation form, the functions are systemically defined as the following:

Secondary Cooling = AFWS
where AFWS = Auxiliary Feedwater System

ECI = HPIS * LPIS + HPRS * LPRS
where ECI = Emergency Coolant Injection
HPIS = High Pressure Injection System
LPIS = Low Pressure Injection System
HPRS = High Pressure Recirculation System
LPRS = Low Pressure Recirculation System

Under each function in the event tree in Figure 5-1, the support systems that will cause the functions to fail are listed. The ICW and CCW systems provide cooling to the ECI system pumps. Failure of the RWST causes failure of the ECI systems. Failure of the diesel generators (DGs) will fail the ECI systems as well as the turbine driven AFWS pumps once battery depletion occurs. The AFWS turbine train failure is assumed to occur either randomly or due to battery depletion in approximately 8 hours.

A sequence resulting in core melt (Sequence 3 in Figure 5-1) requires the failure of the secondary cooling and the emergency cooling injection systems. The probability of this sequence is evaluated in the following section.

0630

TABLE 5-1

VULNERABLE STRUCTURES AND EQUIPMENT

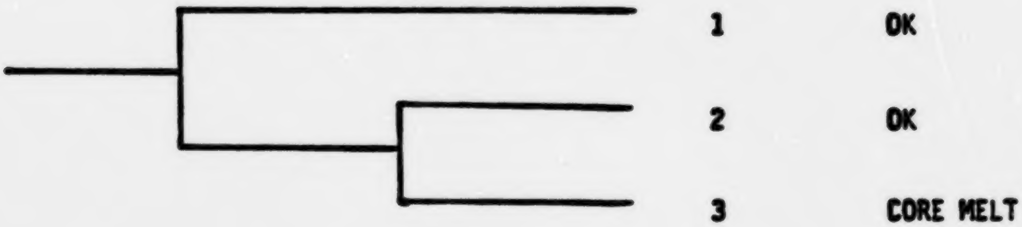
Structure/Equipment	Systems Failed	Comment
CCW Pipes	HPIS/LPIS/HPRS/LPRS	Exposed to tornado missile.
Diesel Oil Storage Tanks	HPIS/LPIS/HPRS/LPRS (1)	Vulnerable to tornado missiles.
RWST	HPIS/LPIS/CSIS	Vulnerable to tornado missiles.
ICW Pipes	HPIS/HPRS/LPIS/LPRS	Exposed to tornado missiles.

---

(1) The AFWS will also fail due to battery depletion.

0631

Function	**Secondary Cooling (SEC)**	**Emergency Coolant Injection (ECI)**	**Sequence Number**	**Core Status**
Vulnerable Support Systems*	DGs**	RWST, CCW, ICW, DGs		



*RWST = Refueling Water Storage Tank
CCW = Component Cooling Water System
ICW = Intake Cooling Water System
DGs = Diesel Generators (Failure due to Loss of DG Oil Tanks)

** The AFWS can fail either randomly or due to battery depletion in eight hours after the DGs fail.

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FIGURE 5-1. WIND/TORNADO FUNCTIONAL EVENT TREE FOR ST. LUCIE

6.0 RISK QUANTIFICATION

Tornado missiles, tornado winds, and straight winds can fail various plant systems resulting in a contribution to core melt probability. In this section the system failure probabilities due to winds and/or missiles are calculated. These system failure probabilities are used in the functional event tree developed previously to determine the contribution to core melt probability.

The methodology used to perform the risk quantification analysis is given in Section 6.1. The results of the analysis for St. Lucie are given in Section 6.2.

6.1 Methodology

The total core melt probability is obtained in the following three steps:

(1) The core melt probability contribution due to straight winds (SW) is calculated, and

(2) The core melt probability due to tornado winds (TW) and tornado missiles (TM) is calculated, and

(3) The total core melt probability is obtained by adding the results from steps (1) and (2) together.

6.1.1 Core Melt Probability Contribution Due to Straight Winds

The core melt probability due to straight winds is obtained in the following steps:

- The probability of failure at each wind speed (fragility) is calculated for each function given in the event tree, Figure 5-1.
- Using the above fragilities the probability of each sequence is determined for each wind speed.

0633

- The frequency of each sequence is obtained by multiplying the probability of each sequence for a given wind speed by the frequency of occurrence of that wind speed. The total sequence probability is obtained by adding the sequence probabilities at different wind speeds together.

- The total probability of core melt is obtained by adding the total sequence frequencies together.

- Recovery actions for each sequence are included whenever appropriate. A new core melt probability (with recovery) is calculated.

To calculate the functional fragilities the following steps are taken:

(1) The equipment/structure fragility is used to determine the system fragilities.

(2) The system fragilities are used to calculate the functional fragilities.

6.1.2 Core Melt Probability Contribution Due to Tornado Winds and Tornado Missiles

The same steps as those used for straight winds in Section 6.1.1 are followed.

6.1.3 Total Core Melt Probability

The total core melt probability is obtained by adding the contributions calculated for straight winds to that calculated for tornado winds and missiles.

6.2 Results

The core melt probabilities are given for straight winds, tornado winds and missiles, and the total (straight winds, tornado winds, and tornado missiles).

6.2.1 Straight Wind Results

No straight wind analysis was required for the St. Lucie plant since the mean frequency of failure for the safety related structures and exposed critical equipment is 1.5E-7 per year (see Section 2.2). Based on previous studies their contribution to core melt frequency would therefore be less than 1E-8.

6.2.2 Tornado Winds and Missile Results

As was the case for straight winds, no analysis was required for tornado winds (see Section 2.2). The tornado missile hazard/fragility for equipment/structures, systems and functions is given in Table 6-1.

The hazard values are obtained from Table 3-5. The equipment/structure failure probabilities are obtained from the fragility parameter values in Table 4-2.

The only failure that would result in core melt (i.e., failure of both the ECI systems and secondary cooling) is failure of the diesel generators (DGs) and battery depletion. Failure of both diesel generators will occur if either of the following occurs: (1) failure of both diesel oil storage tanks due to tornado missiles, or (2) failure of one diesel oil storage tank due to tornado missiles and the other diesel generator that is supplied by the other diesel oil storage tank fails either randomly or due to test or maintenance.

The probability of both diesel oil storage tanks failing due to tornado missiles is the following:

$$P_{Both} = P_{Tornado} \times P_1 \times P_2$$

where P_{Both} = probability that both diesel oil tanks fail due to tornado missiles

$P_{TORNADO}$ = Frequency of occurence per year of any size tornado.

0635

P_1 = probability a missile will hit one diesel oil storage tank given a tornado

P_2 = probability a missile will hit the other diesel oil storage tank given a tornado

P_{Both} = (1.70E-4) x (2.15E-3) x (2.15E-3)

= 7.86E-10

The probability of one diesel oil storage tank failing due to tornado missiles and the diesel generator that is supplied by the other diesel oil storage tank failing either randomly or due to test or maintenance is the following.

$P = P_{Tornado} \times P_1 \times (P_{DG\ (random)} + P_{DG(T\ or\ M)})$

where

P = Probability that one diesel oil storage tank fails due to tornado missiles and the diesel generator that is supplied by the other diesel oil storage tank fails either randomly or due to test or maintenance

$P_{Tornado}$ = Frequency of occurence per year of any size tornado

P_1 = Probability a missile will hit one diesel oil storage tank given a tornado

$P_{DG(random)}$ = Probability that a diesel generator will fail randomly

$P_{DG(T\ or\ M)}$ = Probability that a diesel generator will be unavailable due to test or maintenance

P = (1.70E-4) x (2.15E-3) x (3.8E-2 + 6.1E-3)

= 1.61E-8

The probability of a core melt occurring due to tornado missiles before incorporating any recovery actions is 1.61E-8. Once recovery of offsite power and the possibility of opening the cross connects between the diesel oil storage tanks are included, the core melt frequency will be less then 1E-9. Therefore, the contribution to core melt is insignificant.

6.3 Conclusions and Recommendations

The contribution to core melt frequency due to extreme winds and/or tornadoes is negligible. Even though some of the components at St. Lucie were found to be potentially vulnerable to tornado missiles, their failure did not contribute to the core melt probability. No fixes are needed at St. Lucie for protection from extreme winds.

0637

TABLE 6-1

TORNADO WIND AND MISSILE HAZARD/FRAGILITY FOR EQUIPMENT/STRUCTURES, SYSTEMS AND FUNCTIONS FOR ST. LUCIE

		Failure Probabilities			
Functional Failures**		ECI	ECI	ECI	ECI/Secondary Cooling
Equipment/Structure/System Failures***		RWST	CCW Pipes	ICW Pipes	Diesel Oil Storage Tanks (DGs)
MPH	Frequency*				
66	1.70E-4	1.38E-3(1)	1.07E-2(1)	1.07E-2(1)	2.15E-3(2)
93	9.32E-5				
134	3.31E-5				
182	6.65E-5				
234	1.24E-6				
290	2.03E-7				
349	2.50E-8				

(1) Probability of missile impact.
(2) Probability of a missile impaction on one diesel oil storage tank.
*Frequency of exceedance
**ECI = Emergency Coolant Injection
***RWST = Refueling Water Storage Tank
ICW = Intake Cooling Water System
CCW = Component Cooling Water System
DGs = Diesel Generators

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APPENDIX H

LIGHTNING ANALYSIS FOR THE ST. LUCIE NUCLEAR POWER PLANT

by

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0641

TABLE OF CONTENTS

Page

1.0 INTRODUCTION H-3 643

1.1 Calculation of Lightning Frequency H-3 643

1.2 Calculation of Damaging Strikes at the Site . . . H-6 646

1.3 Evaluation of Plant Protection H-6 646

1.4 Evaluation of Critical Components H-8 648

1.5 Evaluation of Core Melt Probability H-9 649

1.6 Licensee Event Reports (LERs) Related to Lightning H-10 650

REFERENCES H-22 662

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0642

1.0 INTRODUCTION

Each year nuclear power plants report a significant number of lightning-caused turbine trips, plant upsets, and equipment failures. A review of Licensee Event Reports reveals such examples of lightning-caused events as:

- numerous generator trips and reactor trip
- loss of offsite power and 120 V AC vital buses
- loss of a diesel generator power transformer
- blown motor control center fuses
- unavailability of automatic initiation for HPCI and RCIC
- incapacitated annunciators and transformers

Concern resulting from the frequency and consequences of lightning events has led this program to an examination of the probability of lightning leading to a core melt situation. Although a complete analysis of electrical pathways for lightning access to decay heat removal systems is beyond the scope of this program, a bounding calculation of the probability of such an occurrence is attempted and the results can be compared to the plant vulnerability to other special emergencies. The evaluation of plant vulnerability to lightning strikes and subsequent determination of potential contribution to core melt probability proceeds along the following steps:

a) Calculation of the lightning frequency at the site.

b) Calculation of the percentage of damaging strikes.

c) Evaluation of plant lightning protection.

d) Evaluation of critical components.

e) Evaluation of core melt probability.

1.1 Calculation of Lightning Frequency

Conservative estimates indicate that some 100,000 thunderstorms occur within the 48 contiguous states during any one year. It has been observed that flashing rates vary greatly from region to region and also from thunderstorm to thunderstorm. It has also been found that the ratio of cloud-to-cloud and cloud-to-ground flashes vary greatly. However, based on general data, a number of empirical formulae have been developed for interim use to relate cloud-to-ground flash density to thunderstorm-days. For the United States, these relationships are as follows:

Location	Ground Flash Density (km^{-2} yr^{-1})	Reference[1]
Northern U.S.	0.11T	Horn and Ramsey (1951)
Southern U.S.	0.17T	Horn and Ramsey (1951)
U.S.	0.1T	Anderson et al (1968)
U.S.	0.15T	Brown & Whitehead (1969)

T is the number of thunderstorm-days for the locality of interest.

It should be noted that no one relationship can be singled out as the most reliable; however, as a group they provide a general guide to the correspondence between thunderstorm-days and total ground flash density.

A Nuclear Safety Analysis Center study[1] of lightning problems and protection at nuclear power plants has stated that "Relationships developed between lightning activity and latitude are consistent with the observed low incidence of thunderstorms at the poles and the high incidence of thunderstorms at the equatorial belt. However, it has also been observed that the proportion of cloud-to-cloud flashes increases as latitude decreases. It has also been observed that there is a much higher number of flashes in coastal regions than in non-coastal regions. Each of these factors must be considered in an empirical relationship to describe the density of ground flashes per unit time and area for a particular plant site. Pierce, et al, in 1962 proposed the following relationship for cloud-to-ground flash density in terms of latitude and thunderstorm-days per year, T:

$$N_g = (0.1 + 0.35 \text{ Sin latitude}) (0.40 \pm 0.20) T \text{ km}^{-2} \text{yr}^{-1} \qquad \text{Eqn 1}$$

This equation was used to evaluate the relative probability of ground strikes at the various locations investigated."

The U.S. Meteorological Service has charted the number of thunderstorm-days per year within the United States as seen in Figure H-1. For this program, then, we will estimate the number of thunderstorm-days per year for the plant region of concern using Figure H-1. This data will then be applied to Equation 1 to obtain the number of cloud-to-ground flashes per km^2 yr. This number will be compared for accuracy to similar computations using the relationships presented earlier in this section.

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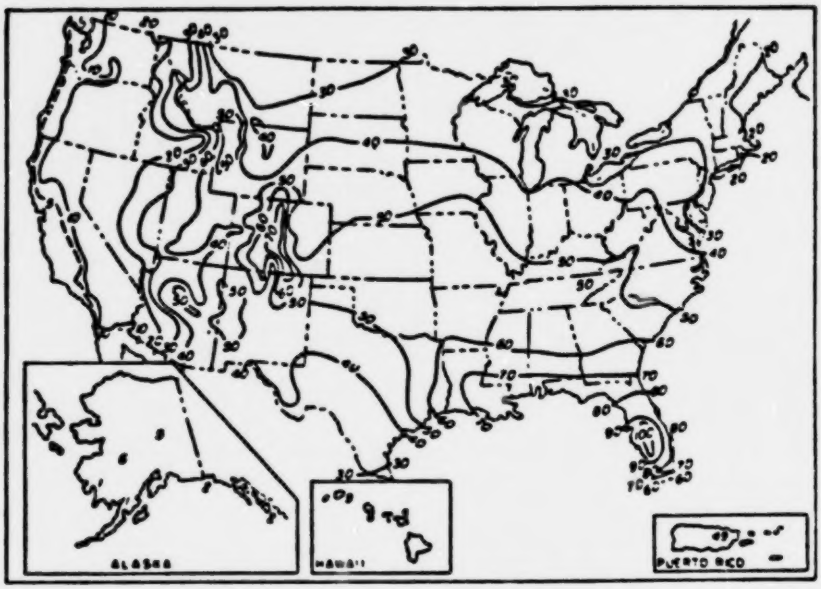

Figure H-1. Map showing the mean annual number of days with thunderstorms. The highest frequency is encountered in south-central Florida. Since 1894, the recording of thunderstorms has been defined as the local calendar day during which thunder was heard. A day with thunderstorms is so recorded regardless of the number occurring on that day. The occurrence of lightning without thunder is not recorded as a thunderstorm. (Data supplied by Environmental Science Service Administration, U.S. Department of Commerce.)

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1.2 Calculation of Damaging Strikes at the Site

NRC recommendations suggest that protection be provided for electrical systems against a lightning strike with a current of 200 kilo-ampere (kA). Available information suggests that one out of every thousand lightning flashes has a current in excess of 200 kA. A recent study by Meier[2] on the frequency of peak lightning current intensities for cloud-to-ground strikes in Florida observed that 1% or less of the flashes observed had currents greater than 100 kA while the largest peak current detected was 418 kA. Figure H-2 gives a cumulative distribution of all strikes as determined by Poplansky in 1972.[1]

Since it is not the purpose of this program to evaluate the effectiveness of existing special emergency protection systems, it will be assumed that a plant with existing protection has met the letter of the NRC recommendations and is not vulnerable to strikes of less than 200 kA. (It should be noted however that the NSAC-41 survey of four nuclear power plants revealed a large number of neglected lightning protection systems. Bent air terminals, insufficient grounding, and corroded conductors are just examples of deficiencies they found in existing protection systems.) This is consistent with the analysis of other special emergencies in which fire sprinklers are assumed to function, seismic snubbers are given credit for successful operation, and watertight doors are assumed to hold tight. The scope of A-45 does not include the exhaustive testing or reanalysis of existing special emergency protective systems.

Therefore, plants with existing lightning protection will be assumed to be protected against lightning strikes of less current than 200 kA. Furthermore, it will be estimated that 1% of all lightning strikes exceed 200 kA regardless of the plant's geographical location.

Using the equations in the previous section to determine the frequency of cloud-to-ground strikes per km^2 yr, and multiplying this number by .01 will yield the number of significant strikes/km^2 yr for a protected site.

1.3 Evaluation of Plant Protection

As previously stated, it is beyond the scope of this program to test or reanalyze lightning protection at nuclear power plants. Rather, this evaluation of plant protection will be limited to a literature review and a plant visit.

The literature review extends only to information available in the public domain. Studies performed by the plant are combined with any details provided in the Final Safety Analysis Report to determine if lightning protection is available at the site. This review is confirmed during a site visit where the building air terminals, ring conductors, and earth shield lines are

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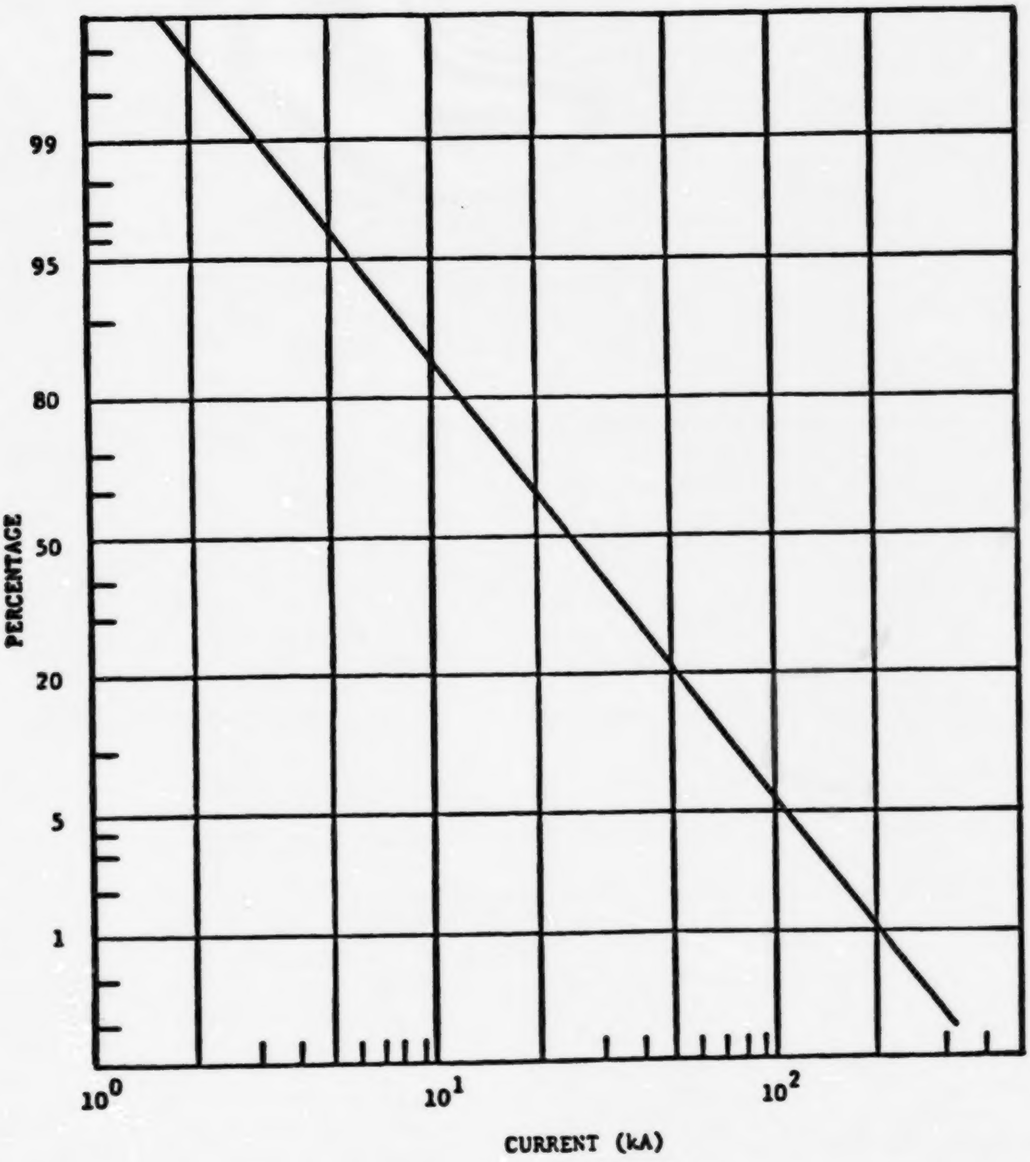

Figure H-2. Cumulative frequency distribution of lightning current amplitudes.

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visually confirmed. Additional questions are asked concerning the ground grid system. If these measures are available, credit is given for lightning protection. It should be noted that lightning has been known to enter buildings via ventilation ducting, roof drains, and roof antennas to name just a few paths. These pathways are too numerous and subtle, however, for this program to evaluate and quantify in depth.

1.4 Evaluation of Critical Components

A review of relevant Licensee Event Reports attributing plant upsets to lightning strikes is provided in this Appendix for detailed review of historical evidence of lightning damage. Examples of system effects include:

- loss of offsite power
- loss of 120 V vital AC buses
- false main steam line isolation and safety injection
- incapacitated annunciators and transformers
- loss of DC power transformers
- failure of battery chargers
- failure of motor control center fuses
- loss of circulating water pump
- failure of charging pumps to start
- unavailability of automatic initiation for HPCI and RCIC
- loss of B train 4160 V emergency bus.

These examples reveal that no system or component can be readily exempted from consideration for lightning vulnerability. The approach, then, is to use the transient event trees from the internal event analysis to determine the success criteria of decay heat removal systems needed following an initiating transient. It is easily conceivable that a loss of offsite power condition will exist due to a storm at the time the lightning strikes the plant. A reactor trip has been the result of many lightning strikes due to such conditions as perturbations in sensors, voltage mismatches, or loss of feedwater pumps. Therefore, the transient event trees are a reasonable source to identify the critical front line and support system combinations necessary for successful decay heat removal.

Although all plants require redundant independent trains of decay heat removal which cannot all be damaged by a single

648

lightning strike, the LERs prove that multiple strikes can hit a plant within a few minutes. Additionally, one train could conceivably be damaged by lightning while the redundant trains could suffer random failure events. Therefore both multiple lightning strikes and single strikes in conjunction with other random failures must be analyzed for their contributions to core melt.

1.5 Evaluation of Core Melt Probability

Once the frequency of lightning strikes with a current greater than the lightning protection design basis has been estimated, and the success criteria of systems needed to respond to a transient have been identified, it is necessary to postulate the probability of a strike damaging one of these essential systems. Given the numerous available lightning targets at a site, as well as the historical evidence of the types of systems that have been hit, it would be considered extremely conservative to assume that one strike out of ten, powerful enough to bypass the lightning protection, will hit a critical decay heat removal system or support system. An example of this would be a pressurized water reactor which would require only auxiliary feedwater or feed and bleed to prevent a core melt. The probability of lightning striking either of these systems, their electric power buses, or motor control centers is small when one considers all possible targets. If a single component, electrical bus, battery, etc., is the necessary target, the probability of a damaging strike, given all of the available targets, is assumed to be one in one hundred (0.01). However, if a single lightning strike could not reasonably fail both systems or trains of electric power, then either a second lightning strike or a random failure would be required to fail the redundant system. As more redundant independent trains are available for decay heat removal, more lightning strikes or random failures would be required for a core melt scenario.

It is probable that plant personnel would effect recovery actions to repair the damage of the lightning strike before core damage could occur. In any case, manual operation of various systems is probable at most plants if power to valves, instrumentation, motor control centers, etc., were to fail. The probability of nonrecovery of damaged systems in time to prevent a core melt is exceedingly remote but will be assumed to be 0.1 as an upper bound for the problem.

The probability of core melt due to lightning would then be calculated by the following equation:

0649

$$\text{Core melt frequency} = T\left[\left(\frac{N_g}{T}\right)(A)(P_{kA})(P_{hit})\right]^{n}(NR) +$$

$$(R_1)T\left[\left(\frac{N_g}{T}\right)(A)(P_{kA})(P_{hit})\right]^{n-1}(NR) + \ldots +$$

$$(R_1)(R_2)\ldots(R_{n-2})(R_{n-1})\;T\left[\left(\frac{N_g}{T}\right)(A)(P_{kA})(P_{hit})\right](NR)$$

where T = Number of thunderstorms-days per year for the region

N_g = Number of lightning strikes per year per km^2

A = Area of the plant site in km^2

P_{kA} = Probability that the lightning current is greater than the design basis.

P_{hit} = Probability that lightning hits a critical component

NR = Probability of nonrecovery (0.1)

n = Number of redundant, independent trains available available to prevent core melt

R_n = Probability of random failure of the n_{th} train

The actual plant specific calculations are straightforward and, therefore, given in Section 3.6 of this report.

1.6 Licensee Event Reports (LER) Related to Lightning

Battery Charger Fails During an Electrical Storm at Vermont Yankee

Description. Date of event - 6/78. Power level - 99%. During a severe electrical storm, control room personnel reported a "Battery Charger CA-1 AC Failure" alarm. Investigation by operating personnel determined that the AC input breaker to the CA-1 battery charger had tripped open. About 15 minutes after the trip condition had occurred, the AC input breaker was reclosed with no further incidents reported. This event was caused by an apparent spurious trip of the AC input breaker on AC&D Company Model No. ARC 130H/K150 battery charger. Cause: spurious trip of input breaker. LER/Docket No. LER 78-002 01T-1 Docket 50-293.

Air Sampler Found Inoperable at Vermont Yankee

Description. Date of event - 6/78. Power level - 99%. During the weekly collection of air particulate samples for the period 6/18-6/25 it was discovered that 3 air samplers had been inoperable for the entire week due to a loose fuse. The 3 air sample pumps stopped operating at about the same time during a severe thunderstorm. Therefore, the blown fuses are attributed to the storm. It is believed that the loose fuse was caused by pump vibration. The blown fuses were replaced and the loose fuse tightened. Cause: fuses blown during thunderstorm. LER/Docket No. 50271.

Stack Gas monitoring System Fails at Vermont Yankee

Description. Date of event - 6/78. Power level - 100%. Due to an impending electrical storm, the stack gas II monitoring system was taken out of service in accordance with the station operating procedures. Subsequently, the operating stack gas I monitor failed due to the lightning strike. The high voltage power supply was repaired and the log count rate board replaced. The stack gas monitors are Model 842-1, manufactured by Victoreen Instrument Division. A plant alternation is being installed that will prevent and/or mitigate the consequences of a lightning strike. Cause: lightning strike. LER/Docket No. 50271.

Power to a Motor Control Center Lost at Vermont Yankee

Description. Date of event - 6/78. Power level - 100%. During a severe electrical storm, control room personnel reported a loss of voltage to motor control center 89B. This MCC powers the "B" LPCI injection valves and the "B" recirculation system discharge and discharge bypass valves. The loss of voltage to MCC 89B was caused by the tripping of an Exide power system 250 KVA uninterruptible power supply. Troubleshooting of the tripped unit did not reveal equipment malfunctions or the cause of the unit trip. Following the energization of the UPS-1B unit, satisfactory valve operability surveillance testing of the "B" LPCI subsystem valves was performed and the unit placed in service. Cause: uninterruptible power supply tripped. LERDocket No. 50-271.

Power to Reserve Transformer Lost at Monticello

Description. Date of event - 6/78. Power level - 60%. During power operations, a lightning storm resulted in plant substation lockouts deenergizing auxiliary reserve transformer 1AR. The relay failed due to short circuit failure of a zener diode in the trip pulse isolator output. This failure is believed to have occurred due to a voltage spike on the 125 VDC supply produced by lightning effects. Zener was replaced with

0651

type 1.5 KE200 having a higher surge rating. Cause: lightning effects. LER/Docket No. LER 78-012 03L-0 Docket 50-263.

All Meteorological Indications Fail at Crystal River 3

Description. Date of event - 7/78. Power level - 0%. All digital and recorded meteorological indications failed contrary to tech spec. Printed circuit cards to the translator power supply and the dew cell were renewed. The cause of this event can be directly attributed to lightning. Cause: lightning. LER Docket No. LER 78-036 03L-0 Docket 50-302.

Loss of All Off-Site Power Scrams the Reactor at Pilgrim 1

Description. Date of event - 8/78. Power level - 100%. A reactor scram occurred as a result of loss of all off-site power. The diesel generators functioned as required and assumed emergency loads, the HPCI and RCIC were manually initiated for water level control and relief/safety valve 203-3C was manually opened once to control pressure and maneuver the reactor to a safe condition. The cause of this event was lightning striking the off-site distribution lines causing a backflash which resulted in high speed relay operation for line-to-ground fault. Power was restored to the startup transformer and station loads returned to normal source after about 1/2 hour. Cause: lightning striking distribution lines. LER/Docket No. LER 78-035 01X-0 Docket 50-293.

Both Roof Vent Flow Recorders Found Reading Downscale at Peach Bottom 3

Description. Date of event - 7/78. Power level - 99%. During power operation the A and B roof vent flow recorders were observed to be reading downscale after an apparent lightning strike. The A monitor recovered without any corrective action within one hour. The cause of the downscale indication was attributed to a lightning strike on the vent stack. The instrumentation was checked and found to be satisfactory except for the indicator drive mechanism on the B flow recorder, which was disengaged. The recorder was repaired, operated satisfactorily and returned to service within ten hours. Cause: lightning strike. LER/Docket No. LER 78-015 03L-0 Docket 50-278.

Lightning Strike Causes Loss of Two Off-Site Power Sources at Prairie Island 1

Description. Date of event - 9/78. Power level - 100%. With both units at full power, a lightning strike on the Spring Creek transmission line caused breaker 6H5 to open. No. 10 bank transformer then tripped on sudden pressure. Two of the three sources of off-site power were now lost and the safeguards buses were now powered from No. 11 cooling tower transformer through bus ties. The diesel generators started automatically and ran unloaded (no loading was required). The second source of off-site power was restored. No. 10 bank transformer should not have tripped on sudden pressure due to the external fault. A design change in transformer protection relaying is being pursued to prevent recurrence. Cause: design deficiency. LER/Docket No. LER 78-018 01T-0 Docket 50-282.

Lightning Strikes Cause Reactor Trip and Safety Injection at Farley 1

Description. Date of event - 6/78. Power level - 100%. Lightning caused 1A circulating water pump to trip. While reducing power, both feed pumps tripped resulting in reactor trip. Lightning caused the supply breakers to S/U transformer 1B to open resulting in loss to B train buses. Loss of 120 V AC vital buses 1C, 1D (control and protection channels III, IV) resulted in SI and main steamline isolation. All D/Gs started, B train D/Gs picked up required buses. Charging pump 1C and RHR pump 1A did not start. Unit shutdown and bus loss resulted from lightning striking near 1A circ water pump and 230 kV switchboard. SI resulted from loss of 120 V AC vital buses 1C, 1D. The cause of the sequencer failure to pick up charging pump 1C, and the failure of RHR pump 1A to start was not determined. Cause: lightning. LER/Docket No. LER 78-033 01T-0 Docket 50-348.

Main Stack Sampling System Lost at Peach Bottom 2

Description. Date of event - 6/79. Power level - 100%. During an electrical storm, both units at power, lightning struck in the vicinity of the main stack. The result was loss of the main stack sampling system, initiation of HPCI and RCIC on Unit 3, and trip of Unit 3 startup and emergency feed. Shutdown of Unit 3 was begun due to unavailability of automatic initiation capability of HPCI and RCIC. Fuse replaced. Shutdown of both units was initiated due to loss of stack sampling system. Shutdowns were terminated upon discovery and replacement of blown fuses. Lightning-induced electrical transient caused blown fuses on stack sample solenoid valve circuit and also on inverter supplying power to HPCI and RCIC initiating instrumentation. Cause: lightning strike. LER/Docket No. LER 79-031 Docket 50-277.

Reactor Scram from Loss of All Off-Site Power at Pilgrim 1

Description. Date of event - 7/79. Power level - 100%. A reactor scram occurred as a result of loss of all off-site power. The diesel generators functioned as required and assumed emergency loads. Relief/safety valve 203-3D was manually actuated to control pressure and maneuver the reactor to a shutdown condition. The cause of this event was lightning

0653

striking the off-site distribution lines causing a backflash that resulted in a high speed relay operation for line-to-ground fault. Power was restored to the start-up transformer and station loads returned to normal source after approximately 1-1/2 hours. Cause: external cause. LER/Docket No. LER 79-027 Docket 50-293.

All Off-Site Power Lost at Pilgrim 1

Description. Date of event - 8/79. Power level - 100%. A reactor scram occurred as a result of a loss of all off-site power. The emergency diesel generators functioned as required and assumed emergency loads. A relief valve was manually actuated to control pressure and maneuver the reactor to a safe shutdown. The cause of this event was lightning striking the station switchyard causing a high speed relay operation for a line to ground fault. Power was restored to the start-up transformer and station loads returned to normal after approximately 30 minutes. Cause: lightning strike in switchyard. LER/Docket No. LER 79-033 Docket 50-293.

Meteorological Instruments Inoperable at Salem 1

Description. Date of event - 10/79. Power level - 000%. During the refueling outage, morning readings showed meteorological instruments inoperable. The required channels were out of service for seven and one-half hours. No atmospheric radioactive releases were made during this time. This is the second occurrence of this type. An apparent lightning strike near the instrument building caused induced current in a raingauge sensor which fed back to the 15 VDC power supply, removing power from all sensors. The damaged electronic components were replaced and the possibility of installing surge protection for all channels is being investigated. Cause: power surge from lightning. LER/Docket No. LER 79-066 Docket 50-272.

Wind Direction Channel Fails Upscale at Browns Ferry 1

Description. Date of event - 1/80. Power level - 000%. While Unit 1 was shut down for refueling, wind direction channel at 33 feet elevation as observed on wind direction recorder XR-90-102-2 failed upscale. Wind direction indicators at 150 feet and 300 feet elevations were inoperable. There have been no previous failures of this nature. The series 312D millivolt-to-current converter was repaired and returned to service in five hours. The indicator failed due to lightning damage at the meteorological tower. Cause: lightning damage. LER/Docket No. LER 80-002 Docket 50-259.

0654

All Off-Site Power Lost at Prairie Island 2

Description. Date of event - 7/80. Power level - 100%. With Unit 2 at 100% power and Unit 1 at cold shutdown, a severe electric storm caused operation of several breakers in the substation. The main generator was separated from the grid and the reactor and reactor coolant pumps were tripped. Natural circulation was established for core cooling. At this point two off-site power sources still existed. No. 10 transformer was locked out and with only one off-site source remaining (a degraded one), the diesel generators started and closed in on the safeguards buses. No safety injection was actuated. Not repetitive. The cause was an electrical storm. The CT11 source was restored and the No. 10 transformer was restored. The next day conditions at the substation were normal. Cause: electrical storm. LER/Docket No. LER 80-020 Docket 50-306.

Off-Site Power Lost at Indian Point 2

Description. Date of event - 6/80. Power level - 100%. During normal plant operation, all off-site electrical power was lost to the site. The reactor was shutdown by the operator via manual turbine trip. The generator was subsequently shutdown by a direct trip signal from the Buchanan substation. The reactor was maintained in a safe shutdown condition using natural circulation cooling until restoration of off-site power. A lightning strike resulted in failure of a system transmission tower shield line which in turn fell and faulted electrical feeders W93,W98,96951 and 96952. The feeders were cleared of the faulted condition and restored to normal service. Cause: lightning. LER/Docket No. LER 80-006 Docket 50-247.

Off-Site Power Lost at Indian Point 3

Description. Date of event - 6/80. Power level - 60%. The unit experienced a loss of outside power on the 138kV feeder. At this point, diesel generators Nos. 32 and 33 were activated to assume the 480V buses. A similar event occurred on July 13, 1977. The cause of the external power loss was determined to be a lightning strike on the ground lead of a transmission tower. The lead became dislodged and fell across the feeder, causing the breaker to open. Repairs were effected and external power was restored. Cause: lightning hit transmission tower. LER/Docket No. LER 80-008 Docket 50-286.

Off-Site Electric Power Line Trips at Browns Ferry 1

Description. Date of event - 8/80. Power level - 73%. Lightning apparently caused Athens 161V line to trip momentarily. After one minute, cooling tower transformer No. 1 tripped causing loss of Unit 3 4kV bus tie board. After about 30 minutes the transformer was reenergized and it tripped again.

Transformer was removed from service for testing and Athens 161kV line reenergized. There were no indications of what tripped the transformer. Cause: lightning strike. LER/Docket No. LER 80-059 Docket 50-259.

EVENTS TAKEN FROM "NUCLEAR POWER EXPERIENCE"

Lightning Strike - Cooling Tower Power Outage, High Blowdown Prairie Island 1 and 2, June 78 - 90% Power

A lightning strike on the plant switchyard caused a power outage to 3 of the 4 cooling towers and a complete loss of power to the blowdown flow control gates. As a result, for a period of approximately 3 hours the circulating water system operated in the helper cycle mode (1 tower operating) with no recirc flow back to the plant.

Meteorological Inverter Switching Mechanism Malfunctioned, Lightning Damaged Power Supply
Salem 1 - Dec 78 and Oct 79 - 100% Power and Shutdown

In December 1978, instrument surveillance at the meteorological tower found that the normal power supply to the inverter had failed and the inverter had not shifted to the battery supply. the inverter was manually shifted to the batteries and proper operation of the instruments was confirmed.

Lightning Burned Out Zeners - Transmission Lines Out
Kewaunee - Aug 73 (hot functional testing)

During a severe lightning storm that struck in the vicinity of their substation, 3 of 4 transmission lines serving the substation tripped due to burn-out of the Zener elements in the Westinghouse relay circuitry. The elements fully meet proposed IEEE standards. Recommended they remove the Zeners from the circuits.

Lightning Strikes, Loss of Off-Site Power - System Blackout, Condenser Rupture Discs Blown
Indian Point 3 - May and July 77 - 91% Power

The 138kV feeder supply outside power to the unit via the station service transformer was lost in May 1977. The loss of outside power was the result of a lightning strike on a feeder in the transmission system. The DGs started and assumed the loads of 480V Buses 5A and 6A. The momentary loss of power to these buses with the resultant loss of NIS Power Range Channel 43 caused a turbine runback to 750 MW. A reactor shutdown was initiated pending evaluation of the incident. After determining the availability of outside power, the shutdown was halted and the unit returned to power. The 138kV feeder was subsequently returned to service.

This occurrence was one of a series of events that led to a blackout of the New York City area. The principal components of the Con Ed system that were damaged consisted of 5 transformers, 10 termination connections, and 15 failed lightning arresters. One of the failed transformers was in the Buchanan Switchyard that provides off-site power for Indian Point 3.

Lightning Caused Loss of Auxiliary Power, Reactor Trip
Cook 1 - Sept 77 - 100% Power

During an extreme thunderstorm, a 345kV transmission breaker faulted and while clearing, lightning hit on a transmission circuit. The protection relaying, to clear the faulted breaker, opened all breakers that could feed to the faulted breaker. This deenergized the transformer from which unit 1 received its normal reserve auxiliary power. Unit 1 was operating at the time, supplying its own auxiliary power, so the loss of the offsite transformer was only the loss of a standby source.

Later, lightning hit and the resultant high-speed opening and reclosure of the associated line breakers on another transmission circuit caused a voltage dip to 64% of normal. The undervoltage protection relays on reactor coolant pump motor buses saw this dip and called for the reactor to trip. The reactor trip cascaded into a turbine trip. By design, the generator trip was delayed 30 seconds after a turbine trip if the initiating trip was a reactor trip. With the generator trip the auxiliary power was automatically transferred from the normal source to the normal reserve source which was deenergized from the earlier incident. Upon loss of voltage, all four 4kV emergency buses shed nonessential loads, the 2 emergency DGs started and accepted their sequential loading within design parameters.

Circulating Water Pump, Start-up Transformer Lost by Lightning Strikes - Scram, Inverter Fuse Failed, SI
Farley 1 - June 78 - 100% Power

Circulating water pump 1A tripped after lightning hit near the pump. The reactor operator initiated a rapid load reduction to prevent a turbine trip due to the rapid increase in condenser backpressure caused by the reduction in circulating water flow. The resultant transient on SG water levels and FW flow caused a low suction pressure trip of both main FW pumps. The reactor tripped when a low FW flow less than steam flow coincident with low SG water level condition occurred. About 18 minutes later, lightning hitting the 230kV switchyard resulted in the loss of start-up transformer 1B when circuit breakers 820 and 924 opened. This caused a loss of power to B Train 416V emergency buses 1G, 1J, 1L, 600V load center 1E, 1J, 1L, 1S, and 416 buses 1B and 1C. All DGs started, and the B train DGs closed on 4160 V emergency buses 1G, 1J, 1L, and accepted load. The Tech Specs required that 4160V emergency buses 1G, 1J, 1L, and 600 V load center 1E, 1J, 1L, 1S be operable, energized and aligned to an operable DG. Transformer 1B was restored to service in approximately 10 minutes.

Lightning Blew Diesel Transformer Fuses
Oyster Creek - June 73

During a plant start-up, with plant electrical loads being supplied by off-site power, a lightning storm occurred in the vicinity of the plant causing an apparent voltage surge on the 1B 4160V bus blowing the 3 power transformer fuses for diesel generator No. 2. The result was to render the fuel oil transfer pumps inoperable since they are among the auxiliaries supplied by the transformer. Although the unit would have started and been operable if a loss of off-site power had occurred, its fuel supply would have been limited to the quantity in the day tank.

Lightning Knocked Out Reserve Transformer - Relay Scheme Modified
Monticello 1 - June 76 - 100% Power

A loss of power to a reserve transformer was experienced. The power loss was due to a lockout of the substation transformer. It was suspected that lightning-induced voltages caused arcing across the sudden pressure protection relay contacts. The relay scheme was modified to use a set of normally closed contacts on the sudden pressure relay to short the coil operated by this relay when not needed.

Lightning Strike Damaged Breaker
Peach Bottom 2 - June 75 - 960 MWe

A lightning strike on the 500 kV system line caused a generator lockout with an accompanying turbine trip and reactor scram. Significant damage was sustained on the 2152 breaker.

Storm-Related Voltage Transients Caused DEH False Signals
Cooper - Aug 80 - 86% Power

A severe weather storm caused a line fault and it was believed subsequent voltage transient signals were induced into the turbine generator Digital Electrical Hydraulic (DEH) system computer. The reactor scrammed automatically when the MSIVs closed.

The DEH computer circuitry sensed false signals indicating the TG output breaker opening. This signal caused the bypass valves to open. With the bypass valves open and the governor valves not controlling the main steam pressure, a Group I isolation occurred due to low pressure and the reactor scrammed on MSIV closure. Further evaluation of the electrical circuitry was planned.

Explosion in Off-Gas System - May Have Been Caused by Lightning
Quad Cities 2 - Mar 73

An OCB in the 345 kV switchyard tripped. The cause was not immediately apparent, however, it was believed to be the result of lightning from a storm in the area. At the same time, the control room operator noticed that the air ejector discharge flow recorder was oscillating in an erratic manner. A power reduction was initiated and then stopped when the flow indication stabilized after 1 minute.

A few minutes later, several area radiation monitors in the turbine building began alarming in the control room and the standby gas treatment system started automatically when its 2 mr/hr setpoint was exceeded. Combined with the other indications the shift engineer was relatively certain that the off-gas rupture disc had blown. The shift foreman and several operators entered the turbine building dressed in protective clothing and full face masks. An inspection of the 2B air ejector room confirmed that the rupture disc had in fact blown and the operators began placing the 2A air ejector in service and isolating the 2B ejector.

Although the exact cause of the detonation was not positively identified, it was believed to have been the result of lightning in the vicinity of the station.

Lightning Caused Explosion in off-Gas System
Vermont Yankee - June 73

While at 78% power with the interim Off-gas (IOG) system in service, and during a severe electrical storm in which a direct lightning strike to the top of the plant ventilation stack was observed, a detonation in the off-gas system caused sufficient back pressure to fracture the air ejector rupture disc, thereby degrading a boundary designed to contain radioactive materials as defined in the tech specs. Concurrently, the area gamma radiation monitor on the perimeter fence became inoperable. This instrumentation is required at all times by the tech specs. The cause of the detonation and instrument malfunction was believed to have been the lightning strike.

Lightning and Possible Smoldering Material Caused 2 Off-Gas System Explosions
Vermont Yankee - Aug & Sept 73

While at 50% power and during a severe electrical storm in which a direct lightning strike to the top of the plant ventilation stack was observed, both plant stack gas monitoring systems and the area gamma radiation monitor on the perimeter fence became

0659

inoperable. Since this instrumentation is required at all times by tech specs, an orderly plant shutdown was initiated.

Lightning Knocked Out Plant, Stack Monitors, Etc.
Vermont Yankee - July 74

At 80% power during a severe electrical storm direct lightning strikes to the top of the plant ventilation stack were observed. Both plant stack monitoring systems became inoperable. About 30 seconds later, another strike resulted in a generator/turbine trip which caused a reactor scram. The perimeter fence monitor, the stack base fan, the process computer, miscellaneous annuciating and indicating circuits, 5 auxiliary equipment transformers, the stack particulate tape transport, the No. 1 fan in the east cooling tower, and the instrumentation from the site meteorological tower were rendered inoperable.

In an effort to prevent the consequence of a direct lightning strike to the plant stack and the problems associated with a plant shutdown without the stack gas monitors in service, they were considering the following:

- Install an additional grounding network at the plant stack that will be connected to the existing plant ground network.
- Revise the applicable plant procedure to provide a means for operating personnel to electrically disarm and disconnect the cables from 1 of 2 stack gas monitors when an electric storm appeared imminent. This would ensure that at least one of the monitors was operable following an electrical storm. This was considered to be an interim measure.
- Install electrical surge suppressors in the circuits of specified control room annunciators.
- In an attempt to remove potential sources of conductance from the stack to the plant buildings, the stack base fan indicating lights be removed from the control room.
- Fabricate a calibration chamber that shields the stack gas detectors from background radiation levels when the detectors are being source calibrated.
- Alter the power source inputs to the monitors in such a manner that one of the monitors may be removed for repair without interrupting the service of the remaining monitors.
- Continue investigation of component failures resulting from severe electrical storms.

The duration of the outage was ≃ 3 days.

Lightning Strike Caused Stack Monitoring System Failure
Vermont Yankee - May 77 - 99% Power

Due to an impending electrical storm, the stack gas II monitoring system was taken out of service. A short time later a direct lightning strike to the top of the plant stack was observed.

Both stack gas monitoring systems failed due to the lightning strike. Stack gas I had a failed log CRM board while stack gas II had a blown control power fuse. The stack gas monitors were Victoreen Model 842-1. Stack gas II was returned to service within 1 hour. An evaluation was to be made of the lightning protection for the control power transformers to the stack monitoring systems.

Lightning Knocked Out Environmental Sample Station
Vermont Yankee - Sept 80 - 98% Power

During weekly environmental air sample collections it was discovered that a continuous sample was not being drawn at Station AT2.3.

The main fuse for the station had blown out ≈8.7 hours into the sampling cycle. A severe electrical storm was reported in the area at that time. A new fuse was installed in the sample station and a subsequent functional check of the station was performed. No further damage was noted. If feasible, lightning surge suppressors were to be installed in all 9 environmental sample stations.

Lightning Damaged Meteorological Transformer and Transceiver
Nine Mile Point 1 - Aug 80 - 97% Power

A loss of meteorological data transmission to control room instruments was experienced for a 10 day period.

While experiencing a rain and electrical storm, a loss of power to the tower equipment occurred from a lightning strike to its power transformer. To prevent recurrence, spare parts were made more readily available.

REFERENCES

1. NSAC-41, Lightning Problems and Protection at Nuclear Power Plants: pg B-13, December 1981.

2. Maier, M.W., A. G. Boulanger, and R. I. Sox, An initial assessment of flash density and peak current characteristics of lightning flashes to ground in South Florida, NUREG/CR-1024, Office of Standards Development, U.S. Nuclear REgulatory Commission, Washington, DC 20555.

APPENDIX J

IMPACT ANALYSIS FOR THE ST. LUCIE NUCLEAR POWER PLANT

by

United Engineers & Constructors, Inc.
Philadelphia, Pennsylvania

CONTENTS

Page No.

1.0 SUMMARY J-4 666

2.0 INTRODUCTION J-7 669

3.0 METHODOLOGY AND APPROACH J-10 672

- 3.1 Design Characterization J-10 672
- 3.2 Operations and Maintenance Assessments J-13 675
- 3.3 Total Evaluated Costs J-13 675
  - 3.3.1 General Approach J-13 675
  - 3.3.2 Economic Ground Rules and Assumptions J-13 675
  - 3.3.3 Capital Costs J-17 679
  - 3.3.4 Replacement Power Costs J-21 683
  - 3.3.5 Operations and Maintenance Costs J-22 684
- 3.4 Radiation Exposure J-23 685

4.0 IMPACT OF PROPOSED ALTERNATIVES J-23 685

- 4.1 Alternative 1 J-23 685
  - 4.1.1 Operations and Maintenance J-23 685
  - 4.1.2 Total Evaluated Costs J-24 686
  - 4.1.3 Radiation Exposure J-29 691
- 4.2 Alternative 2 J-29 691
  - 4.2.1 Operations and Maintenance J-29 691
  - 4.2.2 Total Evaluated Costs J-31 693
  - 4.2.3 Radiation Exposure J-36 698

5.0 CONCLUSIONS J-37 699

6.0 REFERENCES J-40 702

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CONTENTS (Cont'd)

Page No.

ATTACHMENTS

A. Modification Design Report - Condensate and Refueling Water Storage Tank Support J-41 703

B. Modification Design Report - Cable Spreading Room Fire Protection J-47 709

C. Modification Design Report - Alternative Decay Heat Removal Add-on System J-55 717

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PLANT IMPACT ASSESSMENT REPORT

ST. LUCIE DECAY HEAT REMOVAL ALTERNATIVES

1.0 SUMMARY

Two alternative approaches for reducing the vulnerability of the St. Lucie Power Plant to occurrences which could impair its decay heat removal capability were identified by Sandia National Laboratories. In this study, each of these two alternatives was evaluated to determine its impact to the owner-operator in the areas of capital cost, operations and maintenance (O&M) costs, O&M complexity and availability. Since the safety value of each of these approaches was the subject of Sandia work, it was not assessed further here.

In summary, the two alternatives examined are as follows:

Alternative 1 consists of the following:

- Install additional seismic supports on the CST and the RWST.
- Install fire protection enclosures for B Train cable tray located in the Cable Spreading Room.

Alternative 2 is a complete "add-on" system with single trains for both decay heat removal and reactor makeup. Both trains are housed in new buildings with dedicated water supplies and emergency electrical power.

The results of the impact assessments are summarized in Table 1.0-1. All costs are given in real dollars, levelized over the remaining economic

J666

life of the St. Lucie plant. Costs shown are "generic" in nature and are based on data (unit material costs, labor rates, labor productivities) consistent with DOE's "Energy Economic Data Base (EEDB)"; however, the EEDB labor productivities were adjusted to account for known "complicating factors" such as accessiblity, interferences, congestion and the special health/safety considerations inherent in backfit type work.

It is evident from Table 1.0-1 that Alternative 1 which includes minimal new equipment and changes to existing systems or structures, can be achieved at a relatively modest cost and have only a minor impact on plant operations and maintenance. This alternative can also be achieved with minor radiological exposure. The remaining alternative, by adding new structures and a considerable amount of additional equipment, has a major capital cost impact in the range of 50 times that of Alternative 1. The operational and maintenance impact is also much more significant and radiological exposure is higher.

The highest radiological exposure is 20 man-rem for Alternative 2.

Since Alternative 2 involves the addition of active equipment which is required to bring the plant to some prescribed level of safety, it is presumed that the unavailability of this equipment, once installed, will be a limiting condition for operation. Thus, it is expected that this alternative will serve to reduce plant availability somewhat. Although that reduction was not quantified, it is not expected to be large.

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TABLE 1.0-1

ALTERNATIVES IMPACT SUMMARY

ST. LUCIE DECAY HEAT ALTERNATIVES

Alternative	Annualized Capital Costs (Inflated $ x 10^3)	Operations	Maintenance	Radiation Exposure (Man-Rem)
1	196	Minimal-some operator training required but day-to-day operations will not be affected.	Minimal-some additional inspection will be required for the fire protection system but this should be less than 300 mh/yr.	0
2	10,744	Moderate - requires additional line-up checks and significant operator training	Significant - will cost on the order of half a million dollars a year to maintain the added equipment and structures	20

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2.0 INTRODUCTION

The purpose of this report is to provide an assessment of the impact of implementing proposed alternatives for improving the decay heat removal capabilities of the St. Lucie Power Plant. These alternatives have been identified by Sandia National Laboratories as possible approaches to reducing the projected core melt frequency for this plant. This impact assessment, together with similar assessments for other plants, will be used by Sandia as part of the basis for formulating recommendations to the Nuclear Regulatory Commission to resolve Unresolved Safety Issue A-45, "Decay Heat Removal."

A total of two promising approaches, or as they are termed in this report, "alternatives," were identified by Sandia on the basis of probabilistic risk assessment studies. The first alternative consists of two modifications. The second alternative is the addition of a new facility equipped with decay heat removal and reactor makeup systems.

Summary descriptions of each of the modifications proposed for St. Lucie are provided below and are described in somewhat more detail in Attachments A through C. These attachments are presented as Modification Design Reports prepared by the architect-engineer supporting the Sandia work. Each modification was given an identification number for control purposes; these numbers are included here for convenience in referencing individual modifications. The modifications considered are as follows:

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Modification 101 - Condensate and Refueling Water Storage Tank Supports (Attachment A)

This modification reduces plant vulnerability to a seismic event by adding braces to the CST and RWST for Unit 1. Failure of the CST and RWST would result in loss of auxiliary feedwater and safety injection functions respectively. This modification will increase the strength of the tanks sufficiently to ensure their availability following a seismic event.

Modification 802 - Cable Spreading Room Fire Protection System (Attachment B)

The modification reduces plant vulnerability to a fire in the Cable Spreading room by enclosing B Train cable tray that is essential to safety in an intumescent sheet that provides an excellent heat barrier to retard fire transmission. A fire in the Cable Spreading room will therefore be prevented from spreading and becoming a common mode failure for decay heat removal systems.

Modification 812 - Alternative Decay Heat Removal Add-on System (Reference 1, Appendix C)

The purpose of this modification is to improve the reliability of the decay heat removal function by adding additional, protected redundancy. It involves the addition of a new, complete, and dedicated Decay Heat Removal System to assure core cooling and reactor coolant makeup following a loss-of-coolant accident or external events such as fire, flood, tornado and earthquake. One of its subsystems will supply emergency feedwater to the plant's steam generators. Decay heat would be removed by releasing steam through new steam generator atmospheric steam dump valves. The other of its subsystems provides makeup to the Reactor Coolant System in order to maintain inventory. Both of these subsystems are powered from a dedicated diesel and fed from a dedicated water supply.

J670

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This "Add-on" Decay Heat Removal System (ADHR) is housed in two new seismic Category I buildings designed to resist tornado missiles. The ADHR building contains the following components: Alternate Decay Heat Removal Pump, Alternate Service Air Compressor, Alternate Diesel Generator and Associated Tanks, Alternate Makeup Pump, heating and ventilating units, valves, instruments and controls. The other new building contains the Borated Water Storage Tank, Alternate Condensate Storage Tank, Tank Circulation Pumps, valves, instruments and controls.

The Alternate Decay Heat Removal Pump takes suction from the Alternate Condensate Storage Tank and discharges into the existing auxiliary feedwater piping outside containment. The Alternate Makeup Pump takes suction from the new Borated Water Storage Tank and discharges to the safety injection header outside Reactor Containment and then to the Reactor Vessel.

The two alternatives proposed by Sandia are characterized in terms of the individual modifications, which are their constituents, in the following matrix:

UE&C Mod No.	Sandia Mod Designator		Alternative	
			1	2
101	External	1	X	
802	Fire	1	X	
812	Add-on	1		X

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Section 3.0 of this report discusses the methodology and approach which was used to assess the impact that each of the proposed alternatives would have on operations, maintenance, in-service inspection, and radiation exposure and to develop the cost and schedule required for implementation. Section 4.0 presents a discussion of the impact in each of these areas. Section 5.0 provides any general conclusions which can be drawn from this work.

All impact assessments in this report are based on Unit 1 of the St. Lucie Power Plant.

3.0 METHODOLOGY AND APPROACH

3.1 Design Characterization

The objectives of the impact assessment work were to:

- Determine the effect that each of the proposed alternatives would have on subsequent operations, maintenance and in-service inspection.

- Determine all of the costs involved in implementing each alternative. These include not only base capital and replacement power costs needed to install each of the related modifications but also any significant long-term operations, maintenance or in-service inspection costs.

- Determine both the short-term and long-term radiation exposure which will result from implementing each alternative.

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To achieve these objectives it was necessary to first characterize each modification to the extent that such determinations could be made reliably and areas of uncertainty could be identified without a detailed design effort.

This characterization was accomplished in several steps:

- Initial Characterization - Following Sandia's identification of the modifications to be considered, draft Modification Design Reports were prepared. Each report identifies functions and design requirements, describes the proposed design, establishes component design bases, identifies supporting (interface) services required from other systems and structures, and identifies the scope of work (Bill of Materials, rip-out required, etc.) and major construction activities. This first effort was based on available plant documentation which was found to vary greatly in scope, depth and quality with the age and type of the plant and the plant designer.

- Site Visit - Based on the initial characterization, a site inspection plan was prepared for each modification. This plan identified the specific areas which were to be investigated on the site visit in order to confirm the suitability of the design of the proposed modification and to collect cost data and the physical data (congestion, access, radiation level, etc.) needed to permit realistic cost and schedule estimates. Site visits typically involved 2-3 days of intensive effort by site review

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teams consisting of two engineering supervisors (one from each of the two engineering disciplines having the most impact on the design) and one cost estimator each experienced in plant modification work. Data collection involved discussions with plant engineering and operations personnel as well as walkdowns of the systems and areas involved in each modification. In-plant data typically was collected in the form of tape-recorded narrative descriptions and complimentary instant-developing photographs.

- Final Characterization - Data collected on the site visit was used to finalize the Modification Design Reports which appear as appendices to this report. In some cases the initial design approach was simply confirmed; in others, changes were made to correct misconceptions resulting from insufficient initial data or to incorporate worthwhile suggestions made by the plant staff.

This process was changed somewhat for Modification 812 which involved the Add-on Decay Heat Removal System. A previous study done for Sandia (Reference 1) had characterized this system and related structures in some detail. For purposes of this study the Reference 1 design was accepted and used, as is, except that the airplane crash protection features were removed and power supplied from off-site was added. It was felt that, since the remainder of the plant did not include airplane protection features, the added cost was not warranted. No further reviews were done to assess the acceptability of the Reference 1 design.

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3.2 Operations and Maintenance Assessments

Based on the designs, as they are described in the final Design Modification Reports, each of the alternatives was evaluated for its impact on operations and maintenance (including in-service inspection and testing). Since detailed designs were not pursued, full availability models were not available, and funding was limited, these assessments tended to be qualitative rather than quantitative. Thus, they are based on the best judgements of the participating supervising engineers and managers.

3.3 Total Evaluated Costs

3.3.1 General Approach

In developing the cost impact for each modification, the general methodology specified in Reference 2 was used. Specifically, the simplified model approach described in Section 2.4.1 was selected because it provided reasonable accuracy and was more practical in light of the large number of modifications to be considered in the course of the study. The costs incurred by government agencies to specify and assure the implementation of such modifications were not considered.

3.3.2 Economic Ground Rules and Assumptions

To develop total evaluated costs for each alternative, it was necessary to adopt certain economic ground rules and

0675

assumptions. Since it was intended to provide these costs on both a local and generic basis, separate data for both cases has been provided in Table 3.3-1. For a more detailed discussion of the rationale and basis for producing both generic and local estimates, see Section 3.3.3.2 below.

Inflation rate, discount rate, fixed charge rate and O&M levelization factors were estimated using methods and economic data from References 3 and 4.

Replacement power unit costs were estimated using unit costs from Reference 5 which are given by season of the year. Since it is not known when during the year the modifications would be accomplished, these seasonal values were averaged over the year. Also, since Reference 5 costs are given in in January 1984 dollars, they were escalated at 3% to be consistent with the capital cost base.

It should be noted that Reference 5 unit costs are calculated on the assumption that the unit would have been operating at its average annual capacity factor during the period of evaluation. Thus, a day's extension of a shutdown results in replacement power charges for that day equal to the charges for one full power day times the plant's capacity factor. The length of an average refueling shutdown was obtained from reactor suppliers.

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Base O&M labor rates were obtained from the authors of Reference 6. The labor rates shown in Table 3.3-1 are these base rates increased to include the employer's costs for social security and workmen's compensation. The O&M labor/ materials ratio suggested in Reference 3 was used along with total labor to estimate total O&M costs.

Owner's costs are an allowance to account items such as the following which can be directly related to the owners support activities:

- Retraining of Operators
- Engineering support activities such as review and approval of contractor engineering and design work, preparation of procedure and manual changes, interaction with Regulatory agencies, and preparation of as-built drawings.
- Quality Assurance/Control
- Health Physics
- Owner Management

The relationship between owner's costs and capital costs is highly utility and task dependent. Utilities differ on how much they feel they need to be involved in supervising and managing modification tasks, whether they will do any engineering, design, purchasing or construction management themselves and how they choose to account for their activities on a task basis. For this study, it was assumed that an outside agency(ies) would be contracted to do the engineering, design, purchasing and construction management and related

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TABLE 3.3-1

ECONOMIC GROUND RULES AND ASSUMPTIONS

	PARAMETER	GENERIC	LOCAL
1.	Plant Location	Eastern Pennsylvania	Southern Florida
2.	Remaining Economic Life (Yrs)	19 yrs	18 yrs
3.	Inflation Rate (%)	6	6
4.	Discount Rate (%)	10.5	10
5.	Fixed Charge Rate (%)	17.0	16.5
6.	O&M Levelization Factor	1.47	1.63
7.	Replacement Power Costs ($/MW(e) - Day)	550(1)	584
8.	Average Refueling Shutdown Length (Days)		
	PWR	60	60
	BWR	70	--
9.	O&M Labor Rate ($/Hr)(2)	22.80	20.61
10.	O&M Labor/Materials Ratio	40/60	40/60
11.	Owner's Costs (%)	10	10
12.	Contingency (%)	25	25

Note: (1) MAAC Power Pool Average

(2) Includes employer's social security and workmen's compensation contributions

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costs were included in indirect capital costs. Therefore, owner's costs were assumed to be 10% of capital costs to reflect this approach and a utility with average involvement in following contractor work. This allocation for owner's costs is thought to be representative based on Reference 7 which, in turn, is based on an informal survey of Gas-Cooled Reactor Associates member utilities.

3.3.3. Capital Costs

3.3.3.1 Cost Accounts

Capital cost estimates were prepared for each modification and are presented in Section 4.0. In these estimates the DIRECT COST accounts include the following:

- All equipment and materials installed in each modification. Equipment and material quantities are as identified in the Modification Design Reports.
- All labor costs which can be directly accounted for (i.e., not prorated) on a task basis. These include costs for ripout, installation and supporting services such as fire watch, scaffolding erection and dismantling and cleanup. Labor rates are based on direct hire and include social security and workmen's compensation taxes. Where multi-shift work was deemed prudent to keep reasonable crew sizes and avoid extending the shutdown, shift work premiums were added.

The following items are included in the INDIRECT COST accounts:

- Temporary Construction Facilities
- Construction Tools and Equipment
- Local Taxes
- Engineering, Quality Assurance and Construction Management by Design Organization(s)

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- **Field Office, Job Supervision, and QA/QC by the Maintenance Contractor or Constructor**
- **Test and Start-up Engineering**

3.3.3.2 Types of Estimates

Two types of cost estimates were prepared for each alternative. The first, termed a "Quasi-Generic Alternative Estimate," is designed to show the effect that technical/configuration differences would have on costs. In this type of estimate, the costs are all derived from a common cost base using an approach intended to minimize the uncertainties associated with cost estimating techniques. Prepared in this way, these estimates should reflect only the physical differences among different alternatives on a given plant and between the same alternative on different plants. The other type of estimate, termed a "Local Alternative Estimate" is intended to reflect costs which the owner might estimate using a cost data base and techniques more in line with their own local experience. Each of these approaches is discussed in more detail below.

- Quasi-Generic Alternative Estimate - In this type of estimate the component and material specifications and quantities reflect the specific configuration of the alternative and plant for which it has been prepared; however, the cost base from which it is derived does not necessarily reflect the plant owner's experience.

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It was prepared to allow cost comparisons which reflect only technical differences, not differences due to the owner's cost experience or estimating techniques. For that reason, all estimates of this type use cost data consistent with the same generic data base. The generic data base used is the "Energy Economic Data Base (EEDB)" maintained by United Engineers and Constructors for the Department of Energy (Reference 8). The EEDB was chosen as the starting point for these estimates because of its long-term credibility and its known relationship to new plant construction experience in the United States.

In estimating installation direct costs, EEDB material unit costs and labor rates were applied directly; component costs were obtained from vendor quotes, the EEDB or the architect-engineer's in-house data base. However, labor productivities from the EEDB, being representative of new construction work, had to be adjusted to accurately reflect "backfit" work on an operating plant. To facilitate the process of making the necessary adjustments to EEDB labor productivities in a consistent and reliable way, a matrix of labor productivity multipliers was prepared. This matrix, which is based on the architect-engineer's in-house cost data base for backfit work, permitted the estimator to select from a range of multipliers depending on the

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"complicating factors" for that work. These "complicating factors" include things such as congestion, accessibility, interferences, and health considerations such as radiation level, surface contamination levels and temperatures in the work area.

- Local Alternative Estimate - This type of alternative estimate differs from the Quasi-Generic estimate in that the generic data is replaced by data which better reflects the local experience of the owner in the areas of labor rates, productivities and other costs.

For Modification 812 involving the Add-on Decay Heat Removal System, the cost estimate created for Reference 1 was used as a starting point. No attempt was made to verify material takeoffs since detailed, scaled drawings were not available. This estimate required a number of changes to make it plant specific, convert it to the same cost base as the other modification estimates and to eliminate the cost of aircraft crash protection features. The Reference 1 estimate, having been created some time ago, needed to be brought up to a January 1985 cost base and have labor productivities adjusted to reflect current industry experience for backfit work. Also, the estimate was adjusted to account for plant size and physical configuration where they differed from those of the plant used as the basis for Reference 1.

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For all, except Modification 812 involving the Add-on Decay Heat Removal System, it was assumed that the modification work packages would be relatively small and on a short schedule and thus would be paid for from retained earnings. The Add-on Decay Heat Removal System in Modification 812, however, being a much larger single capital investment package with a much longer schedule would most likely incur both escalation and interest charges. Thus, both escalation and interest during construction charges were added to Alternative 2 which consists of Modification 812.

Base capital costs were annualized by multiplying by the appropriate fixed charge rates.

3.3.4 Replacement Power Costs

Total replacement power costs were estimated using estimated replacement power unit costs shown in Table 3.3-1 and the projected additional time, beyond that required for an average refueling shutdown, to complete the modification for each alternative.

The need to extend a refueling shutdown was determined on a case basis. Rough durations for work that needs to be done with the plant shutdown were first developed based on manhours in the first-cut capital cost estimates and reasonable crew sizes. Added costs for multiple-shift work needed to ensure all work is completed within one or consecutive (where practical) shutdowns were then estimated. Then, considering the cost of replacement power, a decision on multiple-shift work was made. Where the shift work differential was small (on the order of a few days of replacement power), shift work would be chosen on the assumption that the plant would opt for shift work early in the job anyway in order to not risk extending the shutdown. For longer periods, the decision was made based on the cheaper option.

Replacement power costs, being one-time charges incurred during the outage, were included in total evaluated costs as capital expenditures and annualized using the appropriate fixed charge rate.

3.3.5 Operations and Maintenance Costs

The added operations and maintenance manhours were estimated for each modification and then accumulated for each alternative. These manhours were converted to levelized annual operations and maintenance costs using labor rates, levelization factors and the operations and maintenance labor/materials ratio shown in Table 3.1-1.

Separate O&M costs were calculated for the generic and local cases.

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3.4 Radiation Exposure

General area radiation level data was obtained for those areas expected to contribute significantly to accumulated dose. This data, gathered either during the site visit or in subsequent telephone calls, was used together with cost estimate manhour data to obtain total man-rem exposure for each modification. Based on a review of the proposed work, judgements were made by the engineers who made the site visit regarding the amount of work that could be prefabricated and what would have to be done in place. The latter, together with the related general area radiation levels, was used directly to obtain the exposure for each modification. The total exposure for an alternative was then obtained by accumulating the doses for each related modification.

Lacking detailed information on radiation levels and not having a detailed work plan, it was not practical to assess whether any temporary shielding would be appropriate. Thus, none was included in the cost estimate. However, there is sufficient contingency to accommodate any shielding should it be required.

4.0 IMPACT OF PROPOSED ALTERNATIVES

4.1 Alternative 1

4.1.1 Operations and Maintenance

The discussion below deals only with impact on operations and maintenance activities which follow full implementation of the modification. Impacts on plant schedules and costs incurred in installing the modification are treated in Section 4.1.2. For purposes of this discussion "Operations

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and Maintenance" also includes in-service inspection (ISI) and periodic testing.

Alternative 1 consists of Modifications 802 and 101. Modifications 101 and 802 do not add any new active components and therefore require no additional maintenance than that required with the existing systems.

It is estimated that Modifications 802 and 101 will however require a total of approximately 300 manhours per year for in-service inspection of the fire barriers and supports.

4.1.2 Total Evaluated Costs

Total evaluated annual costs estimated to result from this alternative are as shown in Tables 4.1-1 and 4.1-3 for the generic and local cases respectively. Associated capital costs showing a breakdown of direct costs by modification and total indirect costs for the entire alternative are shown in Tables 4.1-2 and 4.1-4. Owner's costs and contingency were calculated in accordance with the ground rules discussed in Section 3.3.2. These modifications can be completed in parallel while the plant is on-line or during a single shutdown. There are no replacement power costs. Base capital costs, contingency, owner's costs and replacement power costs were capitalized and fixed charged at rates of 17 and 16.5 percent for the generic and local cases, respectively. O&M costs were calculated using estimated hours identified in Section 4.1.1 and the method described in Section 3.3.5.

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TABLE 4.1-1

TOTAL EVALUATED COSTS FOR

ALTERNATIVE 1 - GENERIC BASIS

(Dollars in Thousands)

1. Base Capital Costs (1/85 dollars)		
Direct Costs	423	
Indirect Costs	322	
Total	745	
2. Contingency	186	
3. Owner's Cost	75	
4. Replacement Power Costs	-	
Total	1,006	
5. Annual Capital Charges		171
6. O&M Costs (1/85 dollars)	17	
7. Levelized Annual O&M Costs		25
8. Total Annual Costs		196

TABLE 4.1-2

CAPITAL COSTS FOR

ALTERNATIVE 1 - GENERIC BASIS

($ in Thousands, 1/85 Cost Base)

Cost Component	Labor		Materials	Total
	Manhour	Amount		
MOD 101	1,000	30	7	37
MOD 802	10,800	300	86	386
Total Direct	11,800	330	93	423
Temporary Constr.			243	243
Engrg. & Constr. Mgt.			79	79
Total Indirect			322	322
TOTAL BASE CAPITAL	11,800	330	415	745

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TABLE 4.1-3

TOTAL EVALUATED COSTS FOR

ALTERNATIVE 1 - LOCAL BASIS

(Dollars in Thousands)

1. Base Capital Costs (1/85 dollars)		
Direct Costs	365	
Indirect Costs	288	
Total	653	
2. Contingency	163	
3. Owner's Cost	65	
4. Replacement Power Costs	-	
Total	881	
5. Annual Capital Charges		145
6. O&M Costs (1/85 dollars)	15	
7. Levelized Annual O&M Costs		24
8. Total Annual Costs		169

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TABLE 4.1-4

CAPITAL COSTS FOR

ALTERNATIVE 1 - LOCAL BASIS

($ in Thousands, 1/85 Cost Base)

Cost Component	Labor		Materials	Total
	Manhour	Amount		
MOD 101	1,000	22	7	29
MOD 802	10,800	250	86	336
Total Direct	11,800	272	93	365
Temporary Constr.			209	209
Engrg. & Constr. Mgt.			79	79
Total Indirect			288	288
TOTAL BASE CAPITAL	11,800	272	381	653

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4.1.3 Radiation Exposure

Installation of the modifications in this alternative will not expose plant personnel to any plant radiation.

4.2 Alternative 2

4.2.1 Operations and Maintenance

Alternative 2 consists entirely of Modification 812 which is the Add-on Decay Heat Removal System. Being a major system housed in its own structures, it is expected to have considerably more impact on operations and maintenance than will any of the other alternatives.

Since the add-on structures will not normally need to be continuously manned by operations personnel, no additional operators are expected to be needed. However, this system will require additional operator training, and additional operator actions during plant operations. The operator will need a good understanding of the interactions of this system with existing decay heat removal trains. In particular, he should know the priority of initiation of the existing and add-on decay heat removal trains and be aware of and be able to interpret indications of each train's performance.

Particular attention will need to be given to this in the design process to ensure that the design of the instrumentation and controls makes the transition to the decay heat removal mode as simple and unambiguous as practical. During

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normal operations, the operator will need to do additional line-up checks to ensure that the system is ready to work in the intended manner when it is called upon to do so.

Considering that the reason for adding this system is to bring the plant to a minimum desired level of safety, it is expected that any failures within the add-on system will constitute a limiting condition for operation. Thus, existence of this system will result in some reduction in overall plant availability.

In-service inspection of all ASME Code, Section III, Class 2 and 3 components in accordance with ASME Code, Section XI will be required. In addition, periodic functional testing of pumps and valves will be required to ensure operability. Also the diesel generator must be tested in accordance with the requirements of Regulatory Guide 1.108, "Periodic Testing of Diesel Generator Units Used as On-site Electrical Power Systems at Nuclear Power Plants." This will require weekly operation at operating temperature as well as monthly operation at full-load conditions.

In addition, the substantial amount of added equipment and structures will require considerable preventive and corrective maintenance as well as general preservation and cleanup activities.

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Operations and maintenance hours were not estimated in detail due to the scope of activities required. Rather O&M costs were estimated as 1% of the direct capital costs which is at the low end of the range of what might be predicted using the methods of References 3 and 6. The low end of the range was selected because some components of cost will not increase (e.g., number of security guards and operators) and the unit is seeing limited duty since it is normally in standby with required operation expected to be infrequent.

4.2.2 Total Evaluated Costs

Total evaluated costs estimated to result from this alternative are as shown in Tables 4.2-1 and 4.2-3 for the generic and local cases, respectively. Associated capital costs showing a breakdown of direct costs by modification and total indirect costs for the entire alternative are shown in Tables 4.2-2 and 4.2-4. Owner's costs, contingency, replacement power, annual capital charges, and O&M costs were calculated as described in Sections 3.3.2 and 4.1.1.

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TABLE 4.2-1

TOTAL EVALUATED COSTS FOR ALTERNATIVE 2 - GENERIC BASIS

(Dollars in Thousands)

1.	Base Capital Costs (1/85 dollars)		
	Direct Costs	20,815	
	Indirect Costs	17,835	
	Total	38,650	
2.	Contingency	9,662	
3.	Owner's Cost	3,865	
4.	Escalation and AFUDC	7,680	
5.	Replacement Power Costs	-	
	Total	59,857	
6.	Annual Capital Charges		10,175
7.	O&M Costs (1/85 dollars)	387	
8.	Levelized Annual O&M Costs		569
9.	Total Annual Costs		10,744

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TABLE 4.2-2

CAPITAL COSTS FOR

ALTERNATIVE 2 - GENERIC BASIS

($ in Thousands, 1/85 Cost Base)

Cost Component	Labor		Materials	Subcontracts	Total
	Manhour	Amount			
MOD 812	412,300	11,168	9,340	307	20,815
Total Direct	412,300	11,168	9,340	307	20,815
Temporary Constr.			11,799		11,799
Engrg. & Constr. Mgt.			6,036		6,036
Total Indirect			17,835		17,835
TOTAL BASE CAPITAL	412,300	11,168	27,175	307	38,650

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TABLE 4.2-3

TOTAL EVALUATED COSTS FOR ALTERNATIVE 2 - LOCAL BASIS

(Dollars in Thousands)

1.	Base Capital Costs (1/85 dollars)		
	Direct Costs	18,762	
	Indirect Costs	16,049	
	Total	34,811	
2.	Contingency	8,703	
3.	Owner's Cost	3,481	
4.	Escalation and AFUDC	6,796	
5.	Replacement Power Costs	-	
	Total	53,791	
6.	Annual Capital Charges		8,876
7.	O&M Costs (1/85 dollars)	348	
8.	Levelized Annual O&M Costs		567
9.	Total Annual Costs		9,443

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TABLE 4.2-4

CAPITAL COSTS FOR

ALTERNATIVE 2 - LOCAL BASIS

($ in Thousands, 1/85 Cost Base)

Cost Component	Labor		Materials	Subcontracts	Total
	Manhour	Amount			
MOD 812	412,300	9,115	9,340	307	18,762
Total Direct	412,300	9,115	9,340	307	18,762
Temporary Constr.			10,608		10,608
Engrg. & Constr. Mgt.			5,441		5,441
Total Indirect			16,049		16,049
TOTAL BASE CAPITAL	412,300	9,115	25,389	307	34,811

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The construction schedule for the add-on system is estimated to be roughly 18 months to 2 years from ground breaking to the the completion of operational testing. It is assumed that at least six months of engineering must be completed before construction could begin. Much of the construction can proceed independent of plant operations; however, tie-ins to the feedwater and Reactor Coolant Systems can only be done with the plant shut down. The tie-ins can be completed in a single 60-day refueling shutdown. Thus, shift differential costs have not been included in the capital cost estimate and there are no replacement power costs.

Escalation and AFUDC costs were computed assuming the bulk of the expenditures would occur during the period January 1986 to June 1987 with a linear cumulative cash flow.

4.2.3 Radiation Exposure

This alternative includes only Modification 812. This modification requires some work in radioactive areas to route piping building into the pipe and valve room to make the tie-in to the safety injection system piping. The total estimated exposure for this work is 20 man-rem.

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5.0 CONCLUSIONS

Table 5.0-1 provides a summary and comparison of the impact assessment for each of the alternatives. This table shows the results for the generic cases only. There is no difference in ranking among alternatives for the generic and local cases, only a reduction in the costs since the local labor rates are lower.

It is evident from the table that Alternative 1, which includes only changes to existing systems or structures, can be achieved at a relatively modest cost and has only a minor impact on plant operations and maintenance. These alternatives can be also achieved with a negligible radiological exposure. The remaining alternative, by adding new structures and a considerable amount of additional equipment, has a major capital cost impact in the range of 50 times that of Alternative 1. Their operational and maintenance impact is also much more significant and their radiological exposure is greater.

Since Alternative 2 involves the addition of some active equipment which is required to bring the plant to some prescribed level of safety, it is presumed that the unavailability of this equipment, once installed, will be a limiting condition for operations. Thus, it is expected that this alternative will serve to reduce plant availability somewhat. Although that reduction was not quantified, it is not expected to be large.

It should be recognized that there is a non-negligible degree of uncertainty in this work. Typically, in modification work, detailed design and a

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TABLE 5.0-1

ALTERNATIVES IMPACT SUMMARY

ST. LUCIE DECAY HEAT ALTERNATIVES

Alternative	Annualized Capital Costs (Inflated $ x 10^3)	Operations	Maintenance	Radiation Exposure (Man-Rem)
1	196	Minimal-some operator training required but day-to-day operations will not be affected.	Minimal-some additional inspection will be required for the fire protection system but this should be less than 300 mh/yr.	0
2	10,744	Moderate - requires additional line-up checks and significant operator training	Significant - will cost on the order of half a million dollars a year to maintain the added equipment and structures	20

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careful system walkdown will identify design features which were not anticipated. Similarly, such an investigation will permit a better feel for application of the "complexity factors" used in cost estimating for the specific modification. Limited budget and time constraints did not permit each modification to be investigated in such detail.

The gap between Alternatives 1 and 2 is so large that these uncertainties would not affect their relative ranking.

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6.0 REFERENCES

1. Study of the Value and Impact of Alternative Decay Heat Removal Concepts for Light Water Reactors, Sandia National Laboratories, D. L. Berry and G. A. Sanders (NUREG/CR-2883, June 1983).

2. A Handbook for Cost Estimating - A Method for Developing Estimates of Cost for Generic Actions for Nuclear Power Plants, Argonne National Laboratory, J. R. Ball, S. Cohen, E. J. Ziegler (NUREG/CR-3971, October 1984).

3. Technical Assessment Guide (TAG), Electric Power Research Institute (EPRI P-2410-SR, May 1982).

4. Moody's Public Utility Manual Moody's Investor Services (1984).

5. Replacement Energy Costs for Nuclear Electricity-Generating Units in the United States, Argonne National Laboratories, J. C. VanKuiken, et.al. (NUREG/CR 4012, October 1984).

6. Non-Fuel Operations and Maintenance Costs for Large Steam-Electric Power Plants, Oak Ridge National Laboratories, M. L. Myers, et.al. (1982)

7. Estimating Ground Rules for the HTGR Program, Gas-Cooled Reactor Associates (GCRA 83-008, February, 1984).

8. Energy Economic Data Base (EEDB) Program, Phase VII Update (1984) by United Engineers & Constructors Inc. for Oak Ridge National Laboratories (to be issued - DOE release pending).

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ATTACHMENT A

MODIFICATION DESIGN REPORT

for

CONDENSATE AND REFUELING WATER STORAGE

TANK SUPPORTS

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CONTENTS

Page No.

1.0 FUNCTION AND DESIGN REQUIREMENT J-43 705

1.1 Function and Purpose of the Modification J-43 705
1.2 Design Requirements and Criteria J-43 705

1.2.2 Performance J-43 705
1.2.1 Safety J-43 705

1.3 Design Codes J-44 706
1.4 References J-44 706

2.0 DESIGN DESCRIPTION J-44 706

2.1 Modification Configuration J-44 706
2.2 Operation/Performance J-44 706
2.3 Controls J-44 706

3.0 COMPONENT DESIGN BASIS J-44 706

4.0 INTERFACE REQUIREMENTS J-44 706

4.1 Support Systems J-44 706
4.2 Structural J-44 706

5.0 SCOPE OF WORK J-45 707

5.1 Installation J-45 707
5.2 Rip-out and Relocation J-45 707

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE J-45 707

6.1 Activities Before Plant Shutdown J-45 707
6.2 Activities After Plant Shutdown J-45 707

--- FIGURE 1 -

DESIGN REPORT

for

CONDENSATE AND REFUELING WATER STORAGE TANK SUPPORTS

1.0 FUNCTION AND DESIGN REQUIREMENTS

1.1 Function and Purpose of Modification

Reduce plant vunerability to a seismic event by providing additional seismic supports for the condensate storage tank (CST) and refueling water storage tank (RWST) of Unit 1.

1.2 Design Requirements and Criteria

1.2.1 Safety

The supports shall be Seismic Category I.

1.2.2 Installation

The CST is located to the west of Unit 1 Turbine Building on grade. The RWST is located to the south east of the Unit 1 Containment Building on grade.

1.3 Design Codes

AISC Specification for the Design, Fabrication and Erection of Structural Steel for Buildings.

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

The CST and RWST are currently anchored by anchor bolts.

In order to increase the strength of both tanks, stiffners will be added as shown in Figure 1 for the RWST. The CST braces will tie into the existing missile shield wall instead of anchoring to ground.

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3.0 COMPONENT DESIGN BASIS

Not applicable

4.0 INTERFACE REQUIREMENTS

None

5.0 SCOPE OF WORK

5.1 Installation

a) Drill bolt holes in concrete support pads

b) Fabricate base support plates

c) Weld support beams to base plates

5.2 Rip-out and Relocation

None

5.3 Bill of Materials

Quantities are given for a total of two tanks

Structural Steel	
15' 4 x 13 wide flange	16
5' 4 x 13 wide flange	16
10 x 10 x 3/4 base plates	32
Drill holes (into concrete pads)	
13/16" diameter	178
Bolts A325	
3/4 x 8" Hilti Standard Kwik Bolts	178

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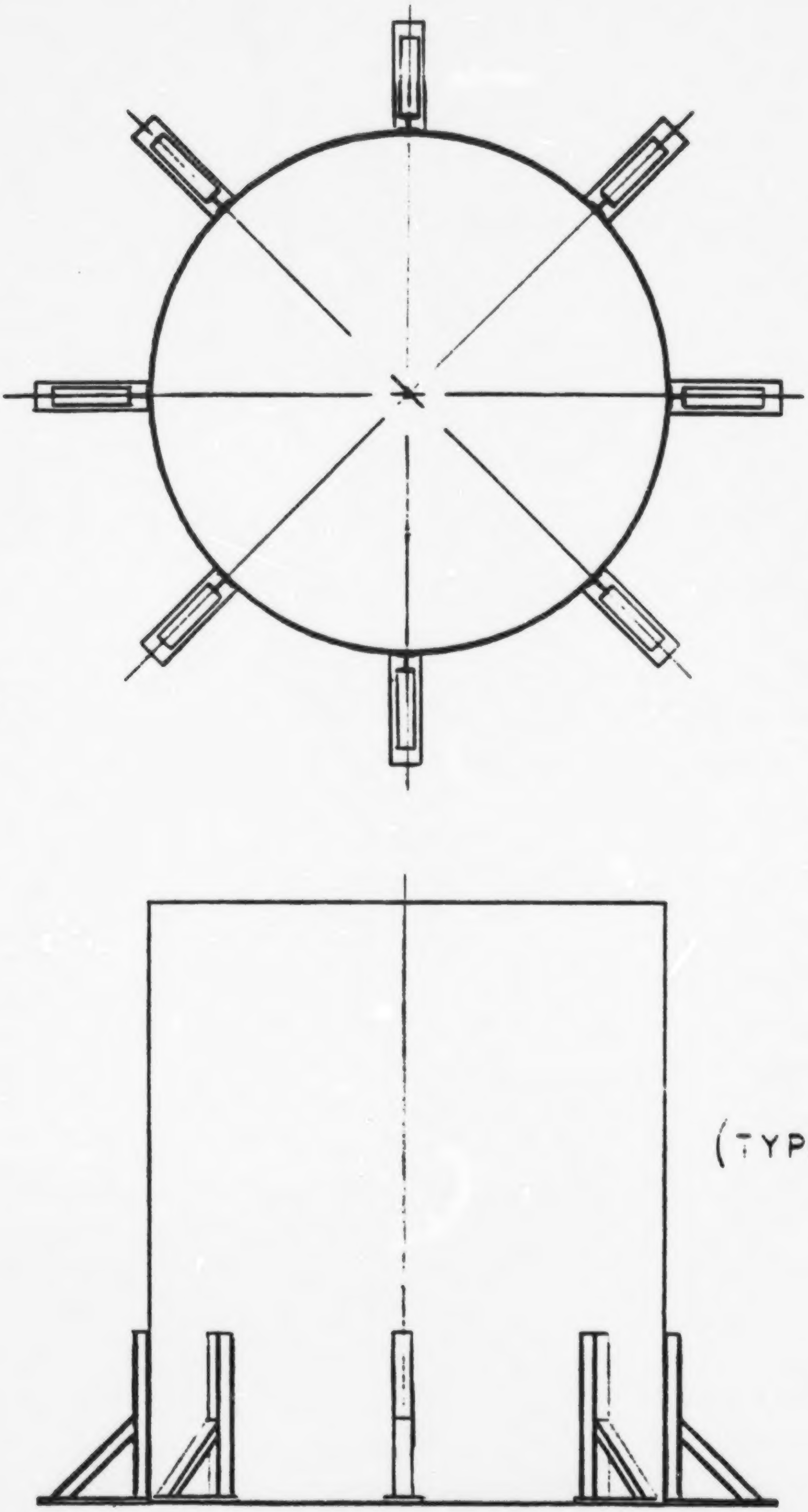
6.0 SEQUENCE OF ACTIVITIES

6.1 Activities before Plant Shutdown

Task				Shutdown
Locate and drill anchor bolts holes	XX			
Fabricate base plates		XX		
Weld wide flanges to base plates and tanks			XXX	

6.2 Activities After Plant Shutdown

None



(TYPICAL CF 4)

FIGURE 1 CST & RWST SEISMIC SUPPORTS

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ATTACHMENT B

MODIFICATION DESIGN REPORT
for the
CABLE SPREADING ROOM
FIRE PROTECTION

CONTENTS

Page No.

1.0 FUNCTION AND DESIGN REQUIREMENT J-49 711

  1.1 Function and Purpose of the Modification J-49 711
  1.2 Design Requirements and Criteria J-49

    1.2.1 Performance J-49 711
    1.2.2 Safety/Regulatory/Availability J-49 711
    1.2.3 Environmental J-49 711
    1.2.4 Installation J-49 711
    1.2.5 Control J-49 711
    1.2.6 Test J-49 711

  1.3 Design Codes J-50 712
  1.4 References J-50 712

2.0 DESIGN DESCRIPTION J-50 712

  2.1 Modification Configuration J-50 712
  2.2 Operation/Performance J-50 712
  2.3 Controls J-50 712

3.0 COMPONENT DESIGN BASIS J-50 712

4.0 INTERFACE REQUIREMENTS J-50 712

5.0 SCOPE OF WORK J-51 713

  5.1 Installation J-51 713
  5.2 Rip-out and Relocation J-51 713

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE J-51 713

  6.1 Activities Before Plant Shutdown J-51 713
  6.2 Activities After Plant Shutdown J-51 713

--- FIGURES 1 & 2 -

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DESIGN REPORT

for

CABLE SPREADING ROOM

FIRE PROTECTION

1.0 FUNCTION AND DESIGN REQUIREMENTS

1.1 Function and Purpose of Modification

The function of this system is to reduce plant vulnerability of fire damage to multiple safety trains due to a fire in the cable spreading room.

1.2 Design Requirements and Criteria

1.2.1 Performance

The modification shall provide an additional 1-hour fire barrier for B Train cable trays in accordance with ASTM E119 requirements.

1.2.2 Safety/Regulatory/Availability

Non-Safety Class
Seismic Category I
Availability - Continuous

1.2.3 Environmental

Location - Indoors
Temperature - 50°F - 100°F
Radiation - Low

1.2.4 Installation

All framing and siding shall be installed in the cable spreading room, suspended from the ceiling wherever possible.

1.2.5 Control

This system is passive and therefore has no controls.

1.2.6 Test

This system is passive and therefore requires no periodic testing.

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1.3 Design Codes

The fire protection material shall conform to ASTM E119 requirements for 1-hour barriers.

The frames shall be designed in accordance with the plant specific seismic requirements.

1.4 References

- St. Lucie Proposed Modifications, 1/6/86, Gary Sanders to Frank Cook
- Ebasco Dwg. No. 8770-G-394, Rev. 30 - Reactor Auxiliary Building El. 43'-0", 62'-0" Conduit, Trays & Grounding, Sheet 1

2.0 DESIGN DESCRIPTION

2.1 Modification Configuration

This modification consists of the addition of a fire protection enclosure which will protect the overhead B Train cable trays against the effects of fire developing in A Train trays in the cable spreading room and vice versa.

The proposed fire protection enclosure system consists of an intumescent sheet product (3M Intram M20A Mat) that when exposed to heat or fire above 250°F rapidly expands into an extremely strong char. This char forms an excellent seal against fire and retards the transmission of heat. Each sheet will be aerated with 1" holes on one-foot centers to permit air circulation during normal operation, thereby avoiding any cable derating (see Figure 1). The sheets shall be supported by and attached to unistrut frames (Figure 2).

2.2 Operation/Performance

This system is passive.

2.3 Controls

Not applicable

3.0 COMPONENT DESIGN BASIS

Intram Mat M20A is tested for surface burning requirements in accordance with ASME E84. The finished installation shall conform to the requirements to 10CFR50, Appendix R.

4.0 INTERFACE REQUIREMENTS

Not applicable

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5.0 SCOPE OF WORK

5.1 Installation

5.1.1 Fabricate and erect 150 unistrut frames 3 ft wide by 35 ft long by 6 ft high over 450 ft of B Train cable tray.

5.1.2 Drill 1-inch holes on 1-foot centers on 375 3 ft by 6 ft M20A sheets

5.1.3 Attach MAT M20A sheets to frames

5.2 Rip-out and Relocation

None. New sheets may be cut as necessary to fit around existing structures, equipment, etc.

6.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

6.1 Activities before Plant Shutdown

6.1.1 Fabricate and install frame support structures

6.1.2 Cut and drill M20A sheets

6.1.3 Attach sheets to frames

		Shutdown
6.1.1	XXXXX	
6.1.2	XXXXX	
6.1.3	XXXXXX	

6.2 Activities after Plant Shutdown

None

0713

TABLE 1

BILL OF MATERIAL

	Quantity
3M MAT M20A Sheets (3' x 6')	375
Unistrut A3300 (6' long)	900
Unistrut AA227 clips	1800
Unitstrut A5300 spans	200
Base Plates (6 x 6 x 3/8)	450
Hilti Quik Bolts (3/8 x 5)	1800
3/4" Self-tapping screws and washers	3600

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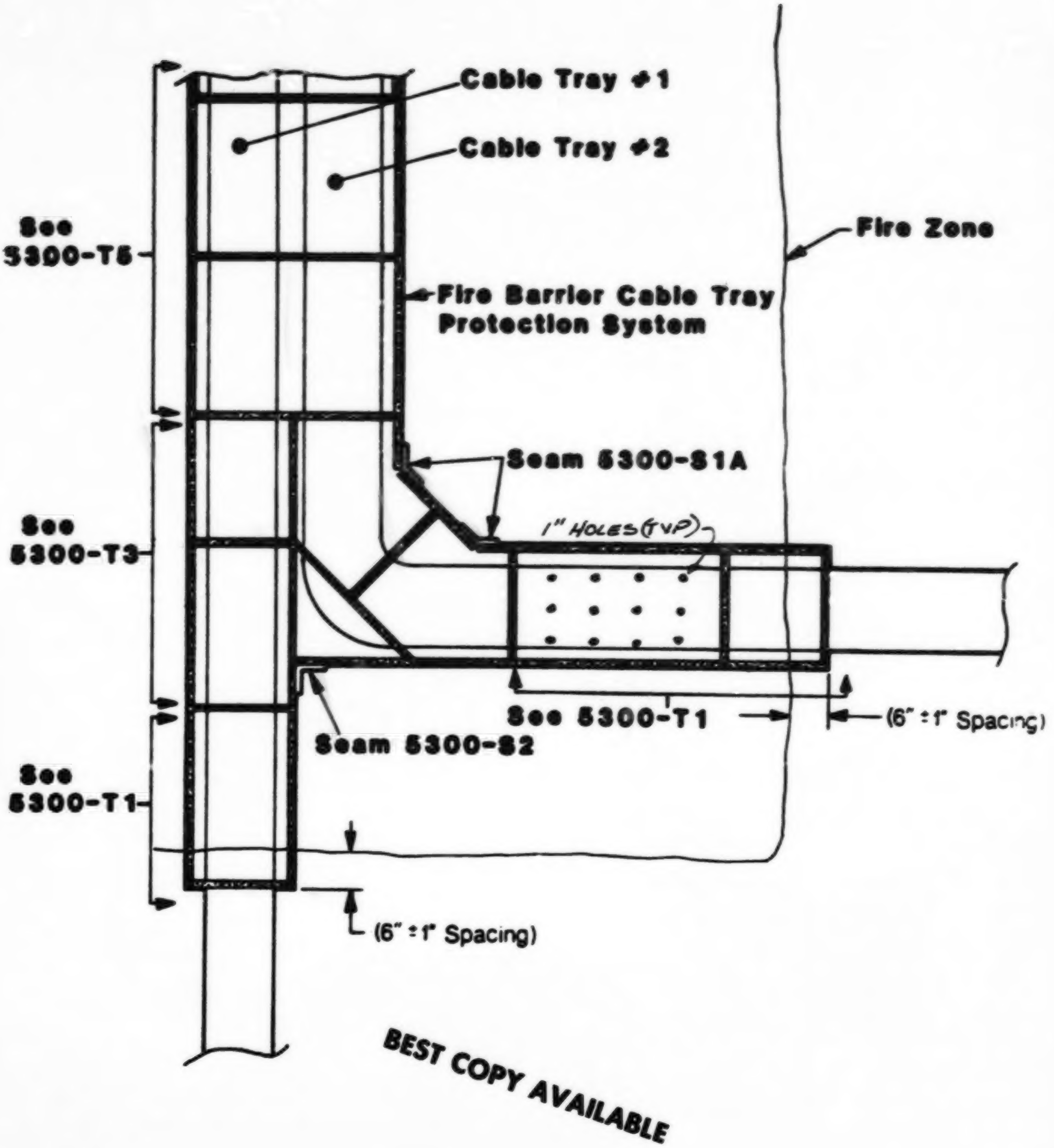
Cable Tray #1
Cable Tray #2
Fire Zone
See 5300-T5
Fire Barrier Cable Tray Protection System
Seam 5300-S1A
1" HOLES (TYP)
See 5300-T3
See 5300-T1
Seam 5300-S2
(6" ± 1" Spacing)
See 5300-T1
(6" ± 1" Spacing)
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FIGURE 1

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Multiple Tray Closure

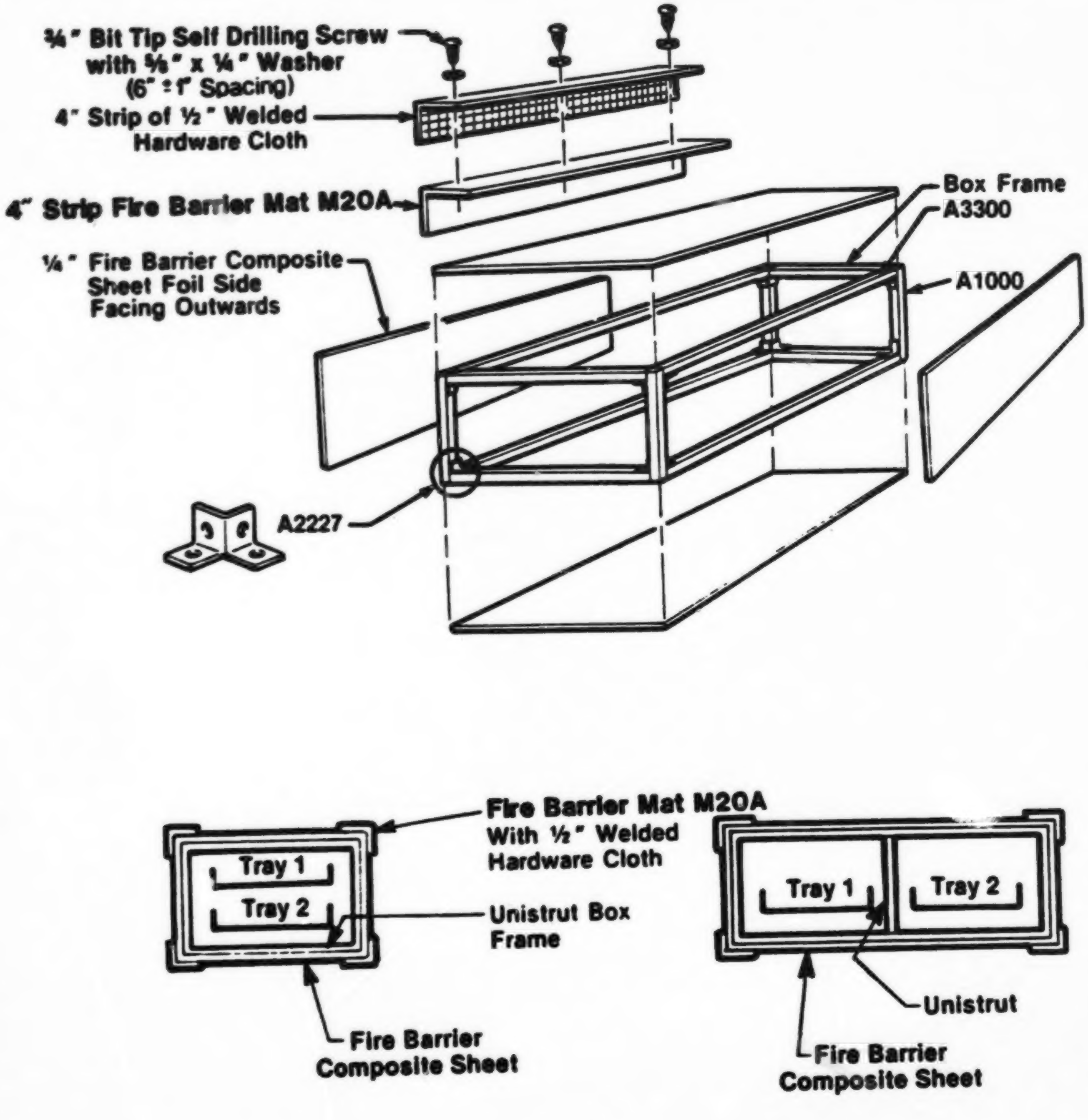
¾" Bit Tip Self Drilling Screw with ⅝" x ¼" Washer (6" ± 1" Spacing)
4" Strip of ½" Welded Hardware Cloth
4" Strip Fire Barrier Mat M20A
¼" Fire Barrier Composite Sheet Foil Side Facing Outwards
Box Frame
A3300
A1000
A2227
Fire Barrier Mat M20A With ½" Welded Hardware Cloth
Tray 1
Tray 2
Unistrut Box Frame
Fire Barrier Composite Sheet
Tray 1
Tray 2
Unistrut
Fire Barrier Composite Sheet

FIGURE 2

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ATTACHMENT C

MODIFICATION DESIGN REPORT
for the
ALTERNATIVE DECAY HEAT REMOVAL
ADD-ON SYSTEM

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CONTENTS

Page No.

1.0 FUNCTION AND DESIGN REQUIREMENTS J-57 719

2.0 MODIFICATION CONFIGURATION AND SCOPE J-57 719

- 2.1 Modification Configuration Description J-57 719
- 2.2 Scope of Work J-58 720
  - 2.2.1 Basis for Outage and Non-outage Work J-58 721
  - 2.2.2 Non-outage Activities J-59 721
  - 2.2.3 Outage Activities J-59 721
  - 2.2.4 Rip-out and Relocation J-59 721

3.0 SEQUENCE OF ACTIVITIES AND SCHEDULE J-60 722

--- FIGURES 1, 2 & 3 -

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DESIGN REPORT

ALTERNATIVE DECAY HEAT REMOVAL

ADD-ON SYSTEM

1.0 FUNCTION AND DESIGN REQUIREMENTS

The function of this system is to provide a new, complete, and dedicated Decay Heat Removal System to assure core cooling and reactor coolant makeup following a loss-of-coolant accident. The design requirements are as described in Reference 1 for Option B.1.

2.0 MODIFICATION CONFIGURATION AND SCOPE

The discussion which follows describes Option B.1 of Reference 1 as it applies to St. Lucie.

2.1 Modification Configuration Description

This modification consists of a single train of feedwater and primary makeup housed within a seismic Category I structure. A schematic diagram is included in Figure 1. Both the feedwater, two 6" supply lines and the primary makeup, one 3" supply line are connected to the existing plant systems outside the containment.

The Add-on Decay Heat Removal System is completely self-contained within the ADHR tank building and ADHR structure. The ADHR tank building houses one 120,000 gal. borated water storage tank for primary makeup and one 200,000 gal. condensate storage tank for auxiliary feedwater. The ADHR tank building and ADHR structure equipment arrangements are included in Figure 2. Both structures are seismic Category I, and in addition to housing the components identified herein, they provide missile protection (turbine and tornado). The tank building is connected to the ADHR structure by a 10-foot seismic Category I tunnel. This ADHR structure houses a single train motor driven auxiliary feedwater pump, a single train motor driven primary makeup pump, a diesel generator complete with all auxiliaries, and the power, control and service systems necessary for operation.

The normal power supply is off-site power via the switchyard and a 34.5/4.16 kV step-down transformer located outside the ADHR structure. Power and control distribution are provided by way of the 4160V and 480V switchgears located within the ADHR structure. The diesel generator is aligned for automatic start and load transfer on the loss of off-site power. Diesel cooling is provided through a close loop service water system arranged to transfer jacket water and lube oil heat to the

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atmosphere via a service water to air heat exchanger (radiator) located within the ADHR. Diesel air intake and exhaust systems are protected from missiles as are the diesel room air cooling system inlet louvers and exhausts.

The ADHR tank building and structure are northeast of Unit No. 1 as shown in Figure 3. The ADHR structure is connected to the existing turbine building feedwater penetration enclosure by a seismic Category I tunnel. This tunnel contains the 3" makeup and two 6" auxiliary feedwater lines in addition to power and control conduit and cable for the motor operated isolation valves steam dump valves and instrumentation.

The tunnels, ADHR tank building to ADHR structure and ADHR structure to feedwater penetration enclosure Unit No. 1, are designed for man access for maintenance and inspection. These tunnels provide missile protection for the piping and conduits.

All piping and conduit within the add-on structures, tunnels, steam trestle enclosure and penetration area are seismically supported with supports located on 10'-0" centers.

The tunnel from the ADHR structure to the steam trestle enclosure connects at ground elevation. All piping there runs through the enclosure to the feedwater piping tie-in locations. The 3" makeup line continues on through the enclosure and around the Reactor Building (encased in a steel chase) and enters the penetration area where it ties into the charging line.

2.2 Scope of Work

2.2.1 Basis for Outage and Non-outage Work

The scope of work is divided in two phases herein. These phases are identified as non-outage work and outage activities.

The non-outage work consists of the ADHR structures and tunnels, pipe, conduit and hanger prefabrication (all areas) and erection of the pipe, conduit, hangers, and components in the new ADHR structures and tunnels and the existing plant auxiliary and turbine building penetration areas.

The outage work consists of pipe, conduit and hanger installation within the feedwater penetration enclosure and Auxiliary Building penetration area and final connections into the existing systems. This scope of work to be done during the outage is balanced between the need to avoid extending the outage and the need to keep radiation exposure to the workers to the minimum.

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0720

The ADHR structures are estimated at an 18-months construction duration. The outage activities are scheduled over a 60-day outage, thereby eliminating the need for replacement power.

2.2.2 Non-outage Activities

These activities begin with site preparation and construction of the ADHR structures and tunnels. All work is complete within the ADHR structures and tunnels, including the new off-site power supply from the switchyard. Following completion of mechanical and electrical system within the ADHR structure and tunnels, the systems are tested. The auxiliary feedwater and primary makeup lines are installed up to the steam trestle and auxiliary building penetration areas.

Piping and conduit in the steam trestle and auxiliary building penetration areas are installed over existing ducts, cable trays and piping to minimize interactions with existing systems and components whose failure could impact the new ADHR system.

In addition to the above, prefabrication of piping conduits and supports for installation during the outage is performed at this time.

2.2.3 Outage Activities

Pipe and conduit supports are installed in the steam trestle enclosure and to the Auxiliary Building penetration area and the final system ties completed.

2.2.4 Rip-out and Relocation

The temporary Security and Records Building located north of Unit 1 will be demolished to allow installation of the ADHR facility.

Temporary relocation of the security fence south of the proposed location for the ADHR facilities will be required to allow construction outside the secure area.

0721

3.0 SEQUENCE OF ACTIVITIES AND SCHEDULE

	Non-outage	|(Start-up)
a.	Site Preparation (including demolition of the Security and Records Building)	XXX
b.	ADHR Tank Bldg. & Structure Substructure	XXXXXX
c.	ADHR Tank Bldg. & Structure Superstructure	XXXXX
d.	ADHR Structure to the Steam Trestle Enclosure	XXXXXXXXXXXX
e.	ADHR Tank Bldg. to ADHR Structure Tunnel	XXX
f.	ADHR Tank Building & Structure Mechanical Equipment Installation	XXXXXX
g.	ADHR Structure Instrumentation Installation	XXXXXXXXXXXXX
h.	ADHR Structure Elec. Equipment Installation	XXXXXXXXXXXXX
i.	ADHR Structure Grounding	XX
j.	ADHR Structure Power Supply	XXXX
k.	ADHR Structure to Steam Trestle Tunnel Piping and Conduit	XXXXXXXXXXXXX
l.	Prefabrication	XXXXXXX XXXXXXXXXXXXX
m.	Piping in Steam Trestle Enclosure	XXXXXXXXXXXXXXXX
n.	Piping from Steam Trestle Enclosure to Aux. Penetration Area (pipe chase)	| XXXXXX
	Outage (Containment & Pipeways)	|Outage
n.	Install Supports	XXX
o.	Install Piping and Conduit	XXXX
p:	Hydro Test	| XXXX

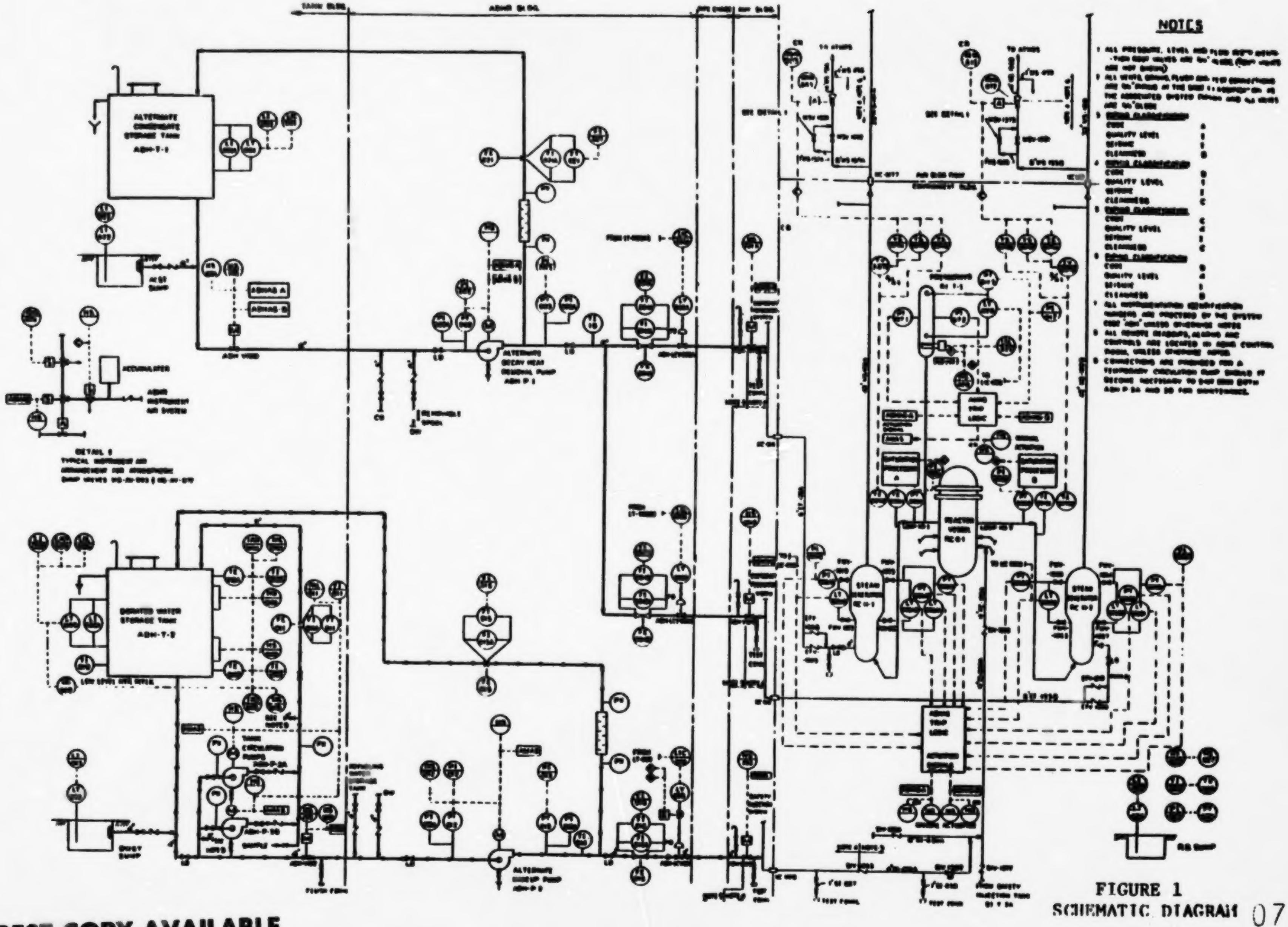
NOTES
ALTERNATE CONDENSATE STORAGE TANK ADH-T-1
BORATED WATER STORAGE TANK ADH-T-2
ALTERNATE DECAY HEAT REMOVAL PUMP ADH-P-1
ALTERNATE MAKEUP PUMP ADH-P-2
ACCUMULATOR
DETAIL 1
PRESSURIZER
REACTOR VESSEL
STEAM GENERATOR
QUALITY LEVEL
CLEANNESS

FIGURE 1
SCHEMATIC DIAGRAM

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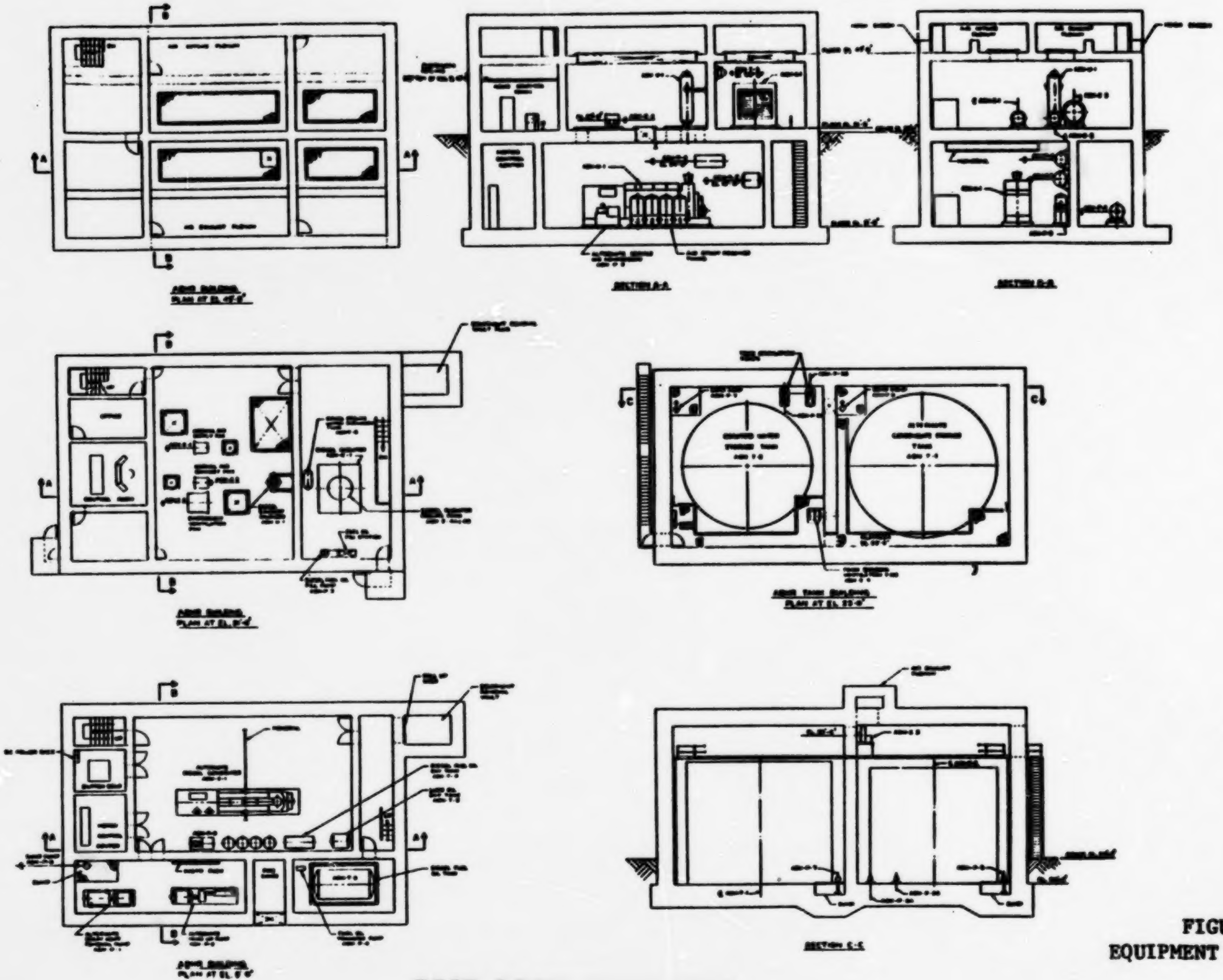

FIGURE 2
EQUIPMENT ARRANGEMENT

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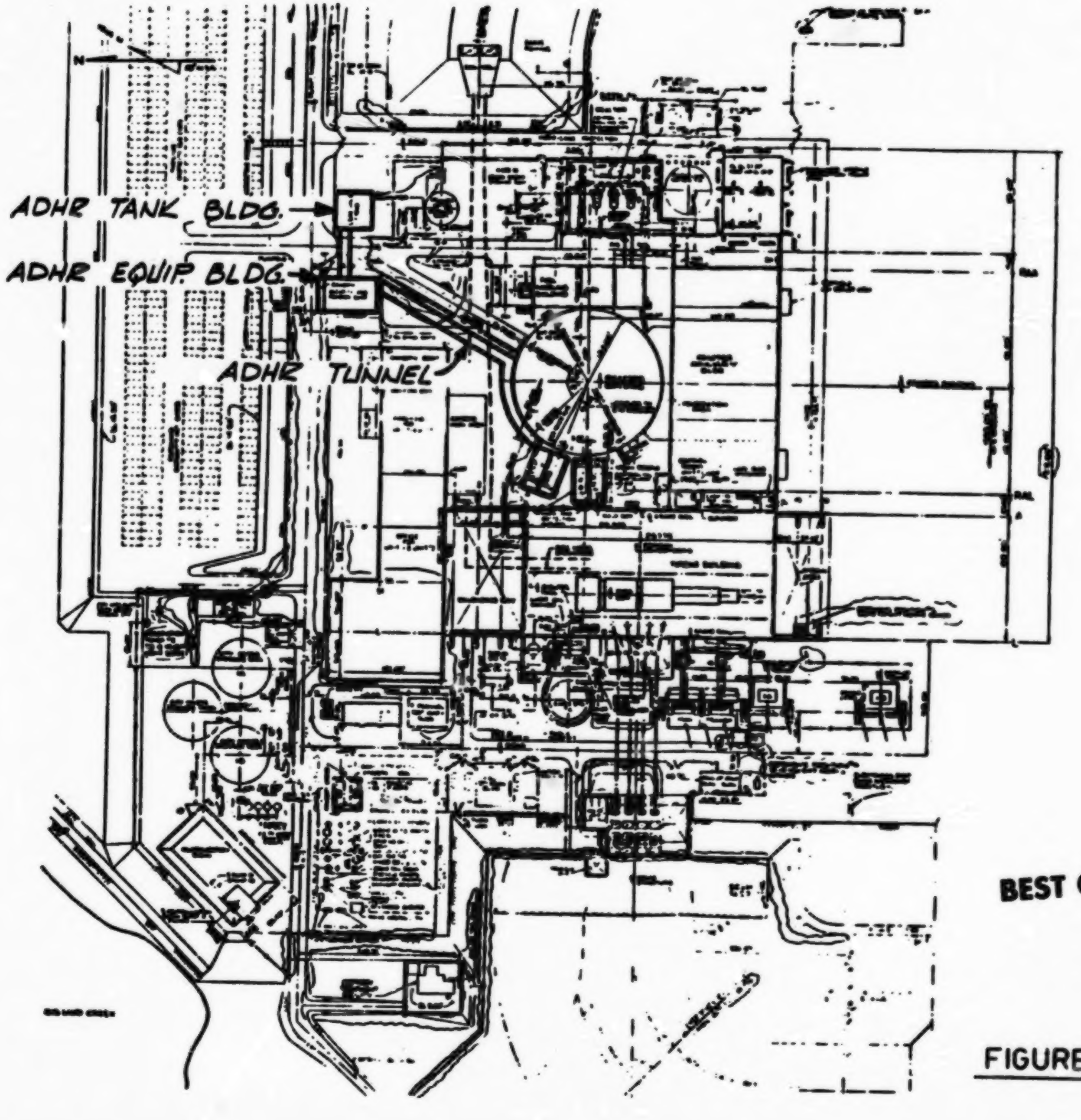
ADHR TANK BLDG.
ADHR EQUIP. BLDG.
ADHR TUNNEL

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FIGURE 3

APPENDIX K

CONSEQUENCE EXTRAPOLATION FOR THE ST. LUCIE NUCLEAR POWER PLANT

by

Gary A. Sanders
Steven W. Hatch
Sandia National Laboratories

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0726

Table of Contents

Section **Page**

1.0 INTRODUCTION K-3 728

2.0 METHODOLOGY AND RESULTS K-3 728

- 2.1 Containment System Modeling K-3 728
- 2.2 Containment Failure Modes and Fission Product Releases K-5 730
- 2.3 Accident Sequence Consequences K-6 731
- 2.4 Use of Consequence Results in Value-Impact Analyses K-10 735

3.0 REFERENCES K-12 737

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0727

1.0 INTRODUCTION

The results of the event tree, fault tree, and accident sequence analyses for St. Lucie include accident sequences, their frequencies per reactor-year, accident sequence cut sets, recovery factors, and proposed plant modifications which would enhance safety. In order to evaluate the consequence to the public and to perform a value-impact assessment on the proposed modifications, a method of estimating the containment response and fission product releases during meltdown accidents was developed.

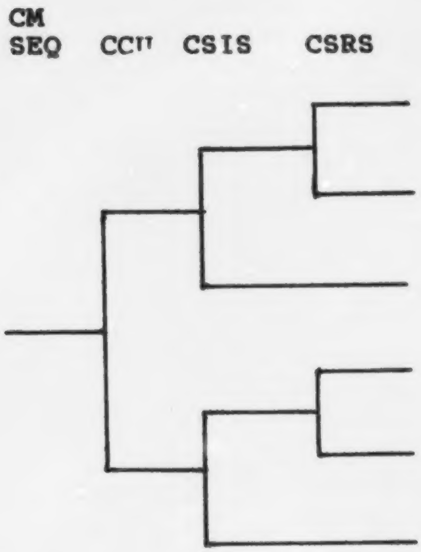
2.0 METHODOLOGY AND RESULTS

The methodology for extending the TAP A-45 PWR results to risk involved three steps: first, fault tree models of containment systems were constructed and containment system status was explicitly included in the accident sequences. This allowed the full spectrum of risk-significant containment failure modes and fission product release calculations from past PRAs to be matched with the important TAP A-45 accident sequences. This matching was based upon several factors which affect containment response during an accident and is discussed below. Next, site-specific consequence calculations were made for each TAP A-45 plant. These calculations estimated the numbers of injuries, fatalities, and amount of property damage that might occur given an accident. Each of the methodology steps is discussed in detail.

2.1 Containment System Modeling

The containment systems do not directly affect the probability of a core melt accident, especially when it is assumed that containment failure prior to core damage will not cause failure of emergency core cooling systems. However, containment system operability during a core melt can influence the ultimate radioactive material release and subsequent exposure, and therefore risk, to the public. Successful operation of containment sprays will lower the radioactive material release by washing out and retaining fission products inside containment. Operation of containment fans or sprays may also prevent or delay containment failure from overpressure. Any delay between initiation of core melting and containment failure provides time for deposition of airborne particles which reduces the potential offsite release.

The containment system event tree used for St. Lucie is presented in Figure 2.1. The sequences delineated by this event tree were combined with the core melt sequences identified by the system event tree (refer to Appendix B, Section 5.4). The containment system event tree delineates the status of two functions: containment overpressure protection (COP) and post accident radioactivity removal (PARR).



Sequence Description

1. COP and PARR always available
2. COP available, no late PARR
3. COP available, no PARR
4. COP and PARR always available
5. COP available, no late PARR
6. No COP or PARR

CM SEQ - Core Melt Sequence

CCU - Containment Cooling Units (Fans)

CSIS - Containment Spray Injection System

CSRS - Containment Spray Recirculation System

COP - Containment Overpressure Protection

PARR - Post Accident Radioactivity Removal

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Figure 2.1 Containment System Event Tree

Successful COP is defined as successful containment fan and/or spray operation throughout the accident to remove decay heat from containment. PARR involves successful operation of containment sprays and is split into two time frames, early and late, to appropriately account for core melts which begin during the injection phase as opposed to the recirculation phase. Knowing the status of COP and PARR during a severe accident is the starting point for estimating containment failure modes and accident releases for St. Lucie.

2.2 Containment Failure Modes and Fission Product Releases

Several probabilistic risk assessments have been published which include calculations of containment response during accidents, calculations of the failure modes and probabilities of containment failure based on this response, and determination of the fraction of core radionuclide inventory which is released from the containment after failure. These calculations are typically generated with computer codes such as MARCH (Reference 1), and CORRAL (Reference 2) and can easily require efforts of a man-year or more to accomplish for one plant according to the PRA Procedures Guide (Reference 3).

In order to assure a timely resolution of TAP A-45 and to accomplish this within reasonable resources it was decided to use existing, published information to estimate the containment failure modes and fission product releases for the TAP A-45 plants. The following paragraphs describe these estimates and the information upon which they were based.

Containment failure modes and fission product release results from the six plants analyzed in the Reactor Safety Study (RSS) (Reference 4) and the Reactor Safety Study Methodology Applications Program (RSSMAP) (Reference 5) were reviewed. These plants and some of their characteristics are listed in Table 2.1.

Table 2.1

Plant	PRA	Vendor	Containment Type
Surry PWR	RSS	West	Subatmospheric
Peach Bottom BWR	RSS	GE	Mark I
Sequoyah PWR	RSSMAP	West	Ice Condenser
Oconee PWR	RSSMAP	BW	Large Dry
Calvert Cliffs PWR	RSSMAP	CE	Large Dry
Grand Gulf BWR	RSSMAP	GE	Mark III

A method of selecting which of the RSS or RSSMAP plants was the best approximation for the TAP A-45 plants in terms of containment response and radioactive material release during accidents was developed. This method involves grouping the plants by containment type and then comparing the ratio of containment failure pressure (P) times containment free air volume (V) to the thermal rating of the plant (MWt). This ratio, PV/MWt, approximately relates the amount of energy the containment can absorb with the amount of energy available. It is expected that plants of the same containment type and similar values of PV/MWt would have similar containment responses for a given accident. It is recognized that the PV/MWt measure is simplistic with substantial uncertainties. For example, this measure does not take into account plant-specific containment and penetration design differences which can affect the results. Based on discussions with containment response experts, however, the method we have chosen is an appropriate first order approximation.

Based on the PV/MWt ratio and containment type, it was concluded that Calvert Cliffs was the closest approximation to St. Lucie in terms of containment response to accidents. However, recent results from other independent probabilistic risk assessments (NUREG-0956; NUREG-1150; IDCOR) have led to the conclusion that the containment failure probabilities used in RSSMAP are unduly conservative. Therefore, based upon a review of available draft reports for NUREG-0956 and NUREG-1150, a revised set of values for use in TAP A-45 when dealing with the large dry PWR containments was generated. The results of the re-analysis which were used for St. Lucie are given in Tables 2.2 and 2.3. Table 2.2 presents the containment failure mode probabilities for each type of core melt accident and their associated fission product release category. Table 2.3 presents the fission product release fraction (fraction of total core inventory which is released) for each of the release categories. Note that the release fractions for a given accident type will vary from plant to plant. Therefore, the St. Lucie results will differ from those of any other plant. This is due to plant-specific design features.

2.3 Accident Sequence Consequences

Using the estimates of the fission product releases described in the last section, the potential consequences that could result from an accident occurring at St. Lucie were calculated. The calculations were performed using the CRAC code (Reference 6). The population distribution around the St. Lucie facility and the local weather patterns were used as input to the CRAC analysis.

The primary CRAC code result used in the St. Lucie analysis is the integrated dose in man-rem received by the population around the plant after an accident integrated out to a distance of fifty miles.

0731

Table 2.2

Identification of Containment Failure Mode Probabilities and Release of Category Assignments for St. Lucie Accidents

Accident Type	Containment Failure Mode Probability	St. Lucie Release Category
1. Core melts with containment sprays and heat removal always available.	$\alpha = 0.0001$	1
	$\epsilon = 0.25$	7
	$\gamma+\delta_e = 0.014$	3
	$\delta_\ell = 0.18$	5
	$\beta = 0.002$	5
2. Core melts with containment heat removal always available. Sprays succeed initially, fail in recirculation.	$\alpha = 0.0001$	1
	$\epsilon = 0.25$	6
	$\gamma+\delta_e = 0.014$	3
	$\delta_\ell = 0.18$	4
	$\beta = 0.002$	5
3. Core melts with containment heat removal available, no containment sprays.	$\alpha = 0.0001$	1
	$\epsilon = 0.25$	6
	$\gamma+\delta_e = 0.014$	2
	$\delta_\ell = 0.18$	3
	$\beta = 0.002$	4
4. Early core melts with no containment heat removal, sprays always available.	$\alpha = 0.0001$	1
	$\epsilon = 0.25$	7
	$\gamma+\delta_e = 0.014$	3
	$\delta_\ell = 0.18$	4
	$\beta = 0.002$	5
5. Core melts with no containment heat removal, sprays succeed early, fail late.	$\alpha = 0.0001$	1
	$\epsilon = 0.25$	6
	$\gamma+\delta_e = 0.014$	3
	$\delta_\ell = 0.18$	4
	$\beta = 0.002$	5
6. Core melts with no containment heat removal, sprays succeed prior to melt, fail at melt.	$\alpha = 0.0001$	1
	$\epsilon = 0.25$	7
	$\gamma+\delta_e = 0.014$	2
	$\delta_\ell = 0.18$	3
	$\beta = 0.002$	4
7. Core melts with no containment heat removal or sprays.	$\alpha = 0.0001$	1
	$\epsilon = 0.25$	6
	$\gamma+\delta_e = 0.014$	2
	$\delta_\ell = 0.18$	3
	$\beta = 0.002$	4

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0732

Table 2.2 (cont.)

Identification of Containment Failure Mode Probabilities and Release of Category Assignments for St. Lucie Accidents

Containment Failure Modes

α	In-vessel steam explosion
γ	Hydrogen burn overpressure
δ_e	Ex-vessel steam spike
β	Containment leakage
ϵ	Base mat melt through
δ_ℓ	Steam and noncondensible gas overpressure

0733

Table 2.3 St. Lucie Source Term

Release Category	* Fission Product Release Fraction						
	Xe	I	Cs	Te	Sa	Ru	La
1	1E-0	7E-1	4E-1	4E-1	5E-2	4E-1	3E-3
2	1E-0	7.5E-1	7.2E-1	6.9E-2	7.9E-2	5.6E-2	9.0E-3
3	1E-0	1.1E-1	1.8E-2	6.0E-1	1.5E-2	3.7E-2	7.1E-3
4	1E-0	9E-2	4E-2	3E-2	5E-3	3E-3	4E-4
5	1E-0	3E-2	9E-3	5E-3	1E-3	6E-4	7E-5
6	1E-0	8E-4	8E-4	1E-3	9E-5	7E-5	1E-5
7	1E-0	2E-5	1E-5	2E-5	1E-6	1E-6	2E-7

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* Based on RSSMAP and RSS Calculations

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Three different sets of CRAC results were calculated. The first calculation uses the release fractions described in the last section and is called the "upper bound" calculation. For the second calculation, all of the release fractions except for the noble gases were uniformly reduced by a factor of seventy percent (0.3 times the upper bound). This is called the "baseline" calculation. For a final "lower bound" calculation, all of the release fractions except noble gases were uniformly reduced by ninety percent (0.1 times the upper bound). These additional calculations were performed to gauge the sensitivity of the potential consequences to variations in the source term. This approach was deemed appropriate in light of the ongoing discussions on reactor accident source terms. The results of the three CRAC code calculations for St. Lucie are given in Table 2.4

It can be noted from Table 2.4 that the 50 mile population doses are relatively insensitive to variations in the source term. This insensitivity occurs because there are competing effects with the early and long-term doses received by the public. The early exposure results from the initial plume passage and is nonlinear because of the threshold and dose rate efects. However, the early exposure is a function of the source term and will decline with it. The long-term exposure results from food and crop contamination and from the assumption that the public will be allowed to reoccupy dcontaminated land at an exposure rate of not more than 25 rem in 30 years. The long-term dose is therefore not as strongly coupled to the magnitude of the source term. In fact, as the source term is reduced, the CRAC code will assume that less land is interdicted and that more land will be decontaminated and reclaimed for public use. This means that the number of people who may potentially receive 25 rem in 30 years by living on decontaminated land may, in fact, increase as the source term is reduced. This factor competes with the reduction in early dose. This effect is magnified by looking at the 50 mile population dose because the decontamination radius is often greater than 50 miles. Because there is so much more land area beyond the decontamination distance in which varying amounts of exposure can occur, the total dose more closely follows changes in the source term.

2.4 Use of Consequence Results in Value-Impact Analyses

The measure of risk which is used in value-impact analysis for St. Lucie is the present worth of the expectation value of radiation dose to the public for the remaining lifetime of the plant. The conditional population doses described in this chapter are used in this calculation. The expectation value of population dose is calculated by summing all the accident sequence probabilities times their containment failure mode probabilities times the appropriate fifty mile dose associated with the containment failure modes. Refer to Appendix L and Section 9 for detailed discussions on the value-impact analysis.

Table 2.4 St. Lucie Population Dose Within 50 Miles

Release Category	Lower Bound (0.1) Source Term	Baseline (0.3) Source Term	Upper Bound (1.0) Source Term
1	5.4E+5	8.5E+5	1.5E+6
2	6.0E+5	8.6E+5	1.4E+6
3	4.5E+5	6.7E+5	1.0E+6
4	2.4E+5	3.9E+5	6.1E+5
5	1.0E+5	2.0E+5	3.6E+5
6	1.5E+4	3.2E+4	7.9E+4
7	4.5E+3	4.8E+3	6.0E+3

0736

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APPENDIX L

METHODOLOGY FOR THE VALUE-IMPACT ANALYSIS FOR THE ST. LUCIE NUCLEAR POWER PLANT STUDY

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0738

CONTENTS

1.0 INTRODUCTION L-3 740

2.0 SPECIFIC NRC REQUIREMENTS FOR VALUE-IMPACT ANALYSES . L-3 740

- 2.1 The Principal NRC Requirements L-3 740
- 2.2 Inplementation of NRC Requirements L-4 741

3.0 METHODOLOGY EMPLOYED FOR VALUE-IMPACT ANALYSIS IN TAP A-45 . . . . L-6 743

- 3.1 Subdivision of the Analysis L-6 743
- 3.2 Methodology of the Impact Analysis L-6 743
  - 3.2.1 The Content of the Impact Analysis . . . L-6 743
  - 3.2.2 Estimation of Engineering Costs L-7 744
  - 3.2.3 Estimation of Other Positive Impacts . . L-8 745
  - 3.2.4 Estimation of "Negative Impacts" Based on Avertable Onsite Costs L-9 746
  - 3.2.5 Possible "Negative Impact" Due to Improvement of Availability L-12 749
  - 3.2.6 Presentation of the Results of the Impact Analysis L-12 749
- 3.3 Methodology of the Value Analysis L-14 751
  - 3.3.1 The Content of the Value Analysis L-14 751
  - 3.3.2 Estimate of Avertable Population Dose . . L-14 751
  - 3.3.3 Estimate of the Avertable On Site Dose. . L-15 752
  - 3.3.4 Estimate of Negative Values Associated with Changes in Occupational Doses . . . L-16 753
  - 3.3.5 Summary of "Value" Terms L-16 753
  - 3.3.6 Special Situations Affecting the Value Analysis L-19 756
- 3.4 Methodology for the Value-Impact Analysis

4.0 REFERENCES L-23 760

1.0 INTRODUCTION

One of the principal objectives of the A-45 program (Reference 1) is to produce a Regulatory Analysis which can serve as a basis for decision-making concerning the need, if any, for improvement of the decay heat removal (DHR) function in existing light water reactors (LWRs). An important part of the Regulatory Analysis is the preparation of "Value-Impact" (or "Benefit-Cost") analyses to determine whether or not proposed modifications are likely to be cost effective.

The NRC has specified certain features which it wishes to see included in Value-Impact analyses and has specified some of the parameter values that should be used (Reference 2). In addition, NRC has sponsored the development of a "Handbook" which sets out methods which should be adopted for these analyses (Reference 3). The Handbook identified three levels of detail (Limited, Intermediate, and Major Effort) at which these analyses could be conducted. The NRC requirements, as set out in Reference 2, are at the Limited Effort level of the Handbook recommendations.

However, both the NRC requirements and the Handbook recommendations leave a number of areas in which the value-impact analyst is left with some freedom of choice. In this Appendix specific recommendations are made as to the methods to be used for TAP A-45.

Most of the recommendations made in this Appendix are at the Limited Effort level of the Handbook but, in order to provide a more adequate treatment of the uncertainties, some extensions to a higher level have been made. Further, it is noted that these recommendations are based on a study of value-impact analysis which has been in progress at UCLA since 1983. Some of the generic results of this study have been reported in References 4, 5, and 6.

This Appendix also provides a description of the formats used to present the results of the value-impact analyses.

2.0 SPECIFIC NRC REQUIREMENTS FOR VALUE-IMPACT ANALYSES

2.1 The Principal NRC Requirements

The basis of the NRC approach to Value-Impact analysis is that the "Value" term should be based primarily on population dose out to a radius of 50 miles from the site, as estimated by the CRAC-2 Code (Reference 7). NRC requires that a value of $1000 per person rem should be attributed to the potential reduction in dose arising from a modification. Value and impact are to be estimated on the basis that the effects of inflation should be disregarded but, with certain exceptions, the reduced worth of payments made or incurred at some future date should be taken into account. For this purpose the effects of using discount rates of 5 percent and 10 percent per annum should be examined. The present worths of quantities expressible in dollars are estimated for the remaining plant lifetime.

Due to the controversy concerning the treatment of the costs arising from material damage to a nuclear power plant (NPP) due to a nuclear accident, the NRC requires that these should be treated as "negative impacts", rather than "positive values", and that they should be shown separately in the analysis. The avertable cost of decontamination and clean up of the NPP and site is also to be treated in this way, except for the dose which would be incurred by personnel engaged in such work; this is to be treated as a "positive value." However, occupational doses incurred in the execution of a modification are to be treated as "negative values;" changes in occupational dose during subsequent operation are also to be treated as a change in the value term.

In the absence of better data, "rule of thumb" figures for cost of clean up and for the associated personnel dose have been proposed by NRC (Reference 2). These are $1.2 billion and 40,000 man rem respectively. A more recent study by Argonne National Laboratory (Reference 8) than that referred to in the PNL Handbook provides data on replacement power costs which are acceptable to NRC (Reference 9).

NRC has also emphasized the need to take account of features peculiar to specific sites, such as additional liquid pathways to humans, and for a careful examination of the uncertainties in the analysis.

2.2 Implementation of NRC Requirements

In the development of the value impact analyses for TAP A-45 the principle has been adopted that in general the analyses should meet the specific requirements stated by NRC in Reference 2 and the detailed procedures set out in the following sections are based on this principle. They are also consistent with the NRC direction that the methods proposed in the Handbook (Reference 3) should be adopted where appropriate. However, as indicated above, to deal adequately with the wide range of uncertainties associated with this type of work requires going beyond a "Limited" analysis and also the development of approaches and methods which are not contained in the Handbook. These departures from the Handbook, have led to recommendations to present the results in a style which differs in some respects, from the examples given in the Handbook.

Table L.1 provides a summary of the required division into "Impacts" and "Values".

0741

Table L.1

Categorization of Attributes into "Impacts" and "Values" for the "Limited Effort" Value-Impact Analysis for St. Lucie

IMPACTS	VALUES
POSITIVE	POSITIVE
Engineering and installation costs of the modifications.	Averted offsite dose
Operations and maintenance costs resulting from the modification	
Replacement power costs during Installation	Averted on site dose due to cleanup and to additional occupational exposure on second unit
Other costs to utility arising from the modification	
NRC costs arising from modification*	
NEGATIVE	NEGATIVE
Averted on site costs (Replacement power, loss of investment, cleanup cost	Additional radiation exposures measured in carrying out modification
POSITIVE OR NEGATIVE	POSITIVE OR NEGATIVE
Changes in plant availability due to Modification (change in outage times for maintenance and in service inspection; changes in outage time due to technical specification)	Change in occupational dose due to modification

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*These costs are not evaluated in the Technical Contractor's report.

0742

3.0 METHODOLOGY EMPLOYED FOR VALUE IMPACT ANALYSIS IN TAP A-45

3.1 Subdivision of the Analysis

The work of performing the analysis is subdivided into five parts as follows:

a) Preparation of the Impact Analysis

b) Preparation of the Value Analysis

c) Examination of the uncertainties in the impact and value analyses

d) Combination of the impact and value analyses to provide a measure of the cost effectiveness of the various alternatives investigated, primarily on the basis of "Best Estimate" values

e) Examination of the effect of the uncertainties on the conclusion reached about the cost effectiveness of the various alternatives using "Best Estimate" data.

The methodology proposed for each of these parts of the value-impact analysis is described below.

3.2 Methodology for the Impact Analysis

3.2.1 The Content of the Impact Analysis

The principal part of the Impact Analysis is the estimation of the engineering costs of executing each of the proposed Alternatives. This is essentially a deterministic process, using methods and data commonly employed in Case Engineering, which is not subject to major uncertainties. However, as in any construction work, unforeseen difficulties may arise, leading to additional cost.

The NRC requirement to treat "avertable on-site costs," as "negative impacts," rather than "positive values," necessitates an extension of the conventional Impact Analysis to cover these items, which are essentially of a probabilistic nature. The NRC also requires that the occupational dose incurred during the installation phase of an Alternative, and any changes in occupational dose during service be treated as "negative values," rather than as impacts.

Another potentially important impact term, in the case of Alternatives which increase the redundancy or diversity of the systems available to perform the DHR function, is the reduction in the outage time that could be incurred in order to comply with the Technical Specifications for the plant in the event of breakdown of equipment in the DHR systems.

0743

3.2.2 Estimation of Engineering Costs

As indicated above, the methods employed for this part of the Impact Analysis are well established and are in constant use in the chemical and other industries; a step towards codifying their application to nuclear power plants (NPP) has recently been taken by the issuance of a "Handbook for Cost Estimating" (Reference 10). A major contribution to the Handbook was made by the Architect-Engineer (AE) who prepared the engineering cost estimates in this program. Thus, as would be expected, the methods and data employed are consistent with those in the Handbook.

In order to reduce the uncertainties in the estimates, the following steps have been taken:

a) Information about local wage rates and the productivity of local labor, relative to the national average values, has been obtained.

b) Visits to the specific plant have been made to confirm impressions gained from the plant drawings. Potential difficulties such as radiation hazards and congestion of working spaces have been discussed with the utility management.

c) The same team of design and cost engineers, experienced in detailed estimating of NPP modifications, has been used to estimate the cost of each of the Alternatives.

The estimates of engineering costs include estimates of the radiation doses which would be incurred during the installation phase and the changes in occupational doses which would be expected during subsequent operation. A monetary value of $1000 is attributed to each additional person-rem which it is expected to be incurred during the installation phase; changes in occupational dose during subsequent operations are monetarized in the same way. Because these changes are relatively certain, as compared with the statistical nature of doses that would be incurred in the event of a reactor accident, they are not discounted. These sums are transferred to the Value Analysis and treated as described in Section 3.3.4 below.

The estimates of engineering costs also include estimates of any plant outage time required for the installation phase of each Alternative. Where it is expected that the required outage time would be greater than that available during a scheduled outage for maintenance or refueling (or a series of such outages), the replacement power cost is estimated using the best available data, which are contained in the recent ANL report (Reference 8) mentioned above. The ANL report provides cost data for each of the seasons, up to Summer 1986. However, because the time at which any Alternative would be put into 0744 effect cannot be forecast at the moment, an average value for the years 1985/1986 has been used. In the case of the St. Lucie Unit 1 NPP the estimated cost is $\$4.48 \times 10^5$ per day.

Possible changes in the materials and equipment costs associated with maintenance and in-service inspection which would arise from each of the proposed Alternatives have also been estimated by the AE. There could also be changes in the outage times required for maintenance and in-service inspection. However, as in the case of possible effects on outage time arising from increased redundancy or diversity of DHR systems (see Section 3.2.1 above), quantification of these times requires additional analysis of system reliabilities. These items, together with possible outages arising from additional spurious scrams or enforced shutdowns, have been excluded from the engineering cost estimates but some comments have been included elsewhere (see Section 3.2.5 below).

3.2.3 Estimation of Other Positive Impacts

In addition to the positive Impacts identified above, there are other quantifiable costs associated with each of the proposed Alternatives; some of these would have to be borne by the utility, others would have to be met by NRC.

The principal additional costs to be met by the utility would be:

a) Changes in labor costs associated with changes in maintenance and in-service inspection requirements.

b) Costs aristing from negotiation with NRC concerning proposed Alternatives.

c) Costs involved in preparation of revised operating procedures and Technical Specifications and in operator training.

d) Cost changes in subsequent operation due to changes in reporting procedures required by NRC (e.g., additional reports on in-service inspections).

It was only possible to prepare an approximate estimate of these costs. Because NRC costs, for development, implementation and operation, due to the proposed Alternatives, can only be established on a generic program basis, it was agreed with the NRC Project Manager for TAP A-45 that these should be included in the overall Regulatory Analysis, rather than in these technical reports for the individual NPP.

3.2.4 Estimation of "Negative Impacts" based on avertable onsite costs.

As indicated in Section 2.1, above, NRC prefers to have most of the avertable on site costs treated as "Negative Impacts" rather than positive values; the avertable on site costs usually included in Value-Impact analyses are as follows:

a) Replacement power costs (i.e., the increase in the cost of generation due to the difference in the cost of nuclear fuel and that of the fuel used in lieu of nuclear, plus any difference in other operating costs).

b) Loss of the remaining value of the utility's investment in an NPP which is not repairable, or cost of repairs and refurbishing.

c) Cost of clean up of plant and site.

d) Radiation exposures resulting from an accident, including cleanup operations.

All of these, except (d), are treated as "Negative Impacts".

There are several other possible causes of substantial losses, which might or might not be borne by the electric power industry: these are discussed in Reference 11.

Since the methods proposed for deriving quantitative estimates for each of Items a) through c) above differ from those described in the PNL Handbook (Reference 3) they are described in detail below.

a) Replacement power costs. It is visualized that it would require 10 years for a utility to replace an NPP in which severe core damage, or core melt, had occurred. After that time it is assumed that power would be provided by a new plant (the capital cost of which would be included in the utility's rate base in the usual way) that at the time would provide the lowest achievable generation cost (i.e., fuel cost + capital cost). Thus the replacement power at any given time is the present worth, at that time, of the differential fuel and operating costs measured in replacing the NPP output, assuming 65 percent capacity factor. At a discount rate of 5 percent, the present worth of 10 years' replacement cost is 7.72 R, where R is the cost for one year (see Table C.2 of Reference 3).

The annual expectation of loss due to potential replacement power costs is 7.72 R p_m, where p_m is the probability of core melt per reactor year. The event can occur at any time during the remaining life of the plant, N years. In a rigorous analysis there should be a correction for the fact that, if core melt were to occur in the last 10 years of life, the

0746

potential loss would be reduced. However, the correction is small and has been omitted here. Thus the present worth in 1985/1986 of the potential loss is the discounted value of N payments of 7.72 Rp_m, at a discount rate of 5 percent.

The St. Lucie Unit 1 has a remaining life of 24 years since it is licensed until the year 2010. Using the value of 24 years for St. Lucie, the present worth of the maximum potential loss is given by Reference 3):

$$PW = 7.72\ Rp_m \left[\frac{(1+r)^N - 1}{r(1+r)^N} \right]$$

where:

r = discount rate = 5%
N = number of years over which loss may occur = 24

Thus for St. Lucie this value is 106.3 Rp_m.

The best available basis for estimating R is provided by the data mentioned in Section 1 above. These are forecasts up to Summer 1986; in the longer term two factors operating in opposite directions could change these:

i) It is generally expected that in the long term the price of fossil fuels will escalate more rapidly than the general inflation rate, whereas nuclear fuel prices will escalate at the general inflation rate.

ii) The proportion of low efficiency and high fuel cost plants on most systems will diminish in the longer term as old plants are retired and are replaced by more efficient plants and as oil-fired plants are converted to coal.

It is assumed, therefore, that the best available estimate can be derived from the average of the last 6 periods given in Table 5.1 of Reference 8. However, the estimate for each period has to be corrected for capacity factor since the tabulated data are based on estimated seasonal capacity factors, with no allowance for planned outages. The average value of 65 percent allows for these outages. On this basis, the annual costs of replacement power (R) for St. Lucie Unit 1 would be \$14.6 x 10^7 per unit.

Thus the maximum avertable replacement power costs, (i.e., if p_m were reduced to zero) would be:

St. Lucie Unit 1 \$1.56 x 10^{10} p_m per unit

Reduction of p_m by an amount Δp_m would give a pro rata saving in replacement power costs.

0747

b) Loss of remaining value of utility's investment. Since all costs and values in the analysis are being estimated in current US dollars, it is considered that the most appropriate method for evaluating the utility's investment at the time of an accident leading to a "write off" is to assume that its original capital cost would have been similar to those of an NPP ordered in 1985/1986, approximately \$1500 per KW(e), and that it depreciates at a uniform rate over a nominal 40 year lifetime.

Thus the estimated value of St. Lucie Unit 1 (802 MW_e) in 1985/1986 would be $802 \times 10^3 \times 1.5 \times 10^3 \times 24/40$, or $\$7.2 \times 10^8$.

The expectation of loss in any one year is $V_c\ p_m$, where V_c is the depreciated value in that year. Thus, the present worth of the maximum expected loss is the sum of the discounted expectations over the remaining life. The maximum avertable costs due to loss of investment value are:

St. Lucie Unit 1 $\$3.7 \times 10^9\ p_m$ per unit

As in (a), above, reduction of p_m by an amount Δp_m would give a pro rata saving in the potential loss of investment value.

The total costs are not greatly different from those which would be obtained using the alternative approach (adopted in Reference 10) that the loss of capital invested can be represented by the total discounted value of replacement power for the remaining life of the plant at the time of the accident; the latter method yields avertable losses which are larger. However, the method used here has the advantage that it shows separately the item which would be the most likely to be passed on directly to the consumer, i.e., the replacement power costs for the first 10 years after an accident.

(c) Cost of plant and site cleanup. In the absence of any better data, the NRC "rule of thumb" figure of $\$1.2 \times 10^9$ is used. It is assumed that this represents the present worth, at the time of the accident, of expenditures that would be spread over several years.

The present worth in 1985/1986 is found in the same way as for the replacement power cost. Thus, at a 5 percent discount rate, the maximum avertable costs for cleanup are:

St. Lucie Unit 1 $1.2 \times 10^9 \times 13.8\ p_m$ or $\$16.6 \times 10^9\ p_m$

Since the accident sequence analysis is based on core melt no attempt has been made to estimate the cost of repairing and refurbishing an NPP after a less severe accident.

0748

(d) Total avertable on site costs. Summing the three components estimated above, the maximum total avertable on site costs are:

St. Lucie Unit 1 $\$3.59 \times 10^{10}\ p_m$ per unit

To obtain the savings for a reduction of Δp_m in p_m, these amounts are multiplied by $\Delta p_m/p_m$.

3.2.5 Possible Negative Impact Due to Improvement of Availability

As indicated in Sections 3.2.1 and 3.2.2 above, a major modification to improve the DHR function may affect availability favorably by reducing the frequency of enforced outages, due to an improved capability for meeting Technical Specifications requirement when repairs to DHR related plant items are in progress. However, adverse effects on availability are also possible due to an increase in the frequency of spurious reactor trips or an increase in the time required for maintenance.

An indication of the extent of the "Negative Impact" is provided by the following illustrative example: It is supposed that a modification avoids the need for an enforced shutdown to meet the Technical Specification once in three years and that on each occasion an outage of 3 days is avoided. Thus, the average saving would be 1 day per year. Taking an average value of $\$0.5 \times 10^6$ per day for the cost of replacement power, the present worth of this saving over 25 years would be $\$7 \times 10^6$.

Since a major modification such as an "add-on" auxiliary feed or HPIS train would be operated in a standby role, the effect on the spurious trip role should be negligible. With care in design the additional outages for maintenance should also be negligible.

3.2.6 Presentation of the Results of the Impact Analysis

Detailed results of the estimation of the engineering costs and of the utility's additional costs are presented in tables similar to those suggested in the PNL Handbook (Tables 3.7 and 3.8 of Reference 3). An overall summary of the Impact Analysis, covering all of the Alternatives, is presented in the format shown in Table L.2. This table includes also upper and lower bounds for the avertable on site costs, to take account of the likely error in the estimates of the probability of core melt (p_m). Due to this inclusion of "Special Emergencies" as initiating events, it is considered that a factor of 5 on either side of the "Best Estimate" (or Point) value is required to provide upper and lower confidence limits.

0749

TABLE L.2 ST. LUCIE UNIT 1 - SUMMARY OF IMPACTS (Based Upon 5% Discount)

		POSITIVE IMPACTS ASSOCIATED WITH MODIFICATIONS (Present Worths)					NEGATIVE IMPACTS DUE TO AVERTABLE ONSITE COSTS (Present Worths)				
		Utility Costs		Replacement Power Costs							
Alternative No.	Change in Core Melt Probability [Central Value]	Installation and Engineering Costs ($ x10^{-6})	Operations and Maintenance Costs (PW) ($ x10^{-6})	Installation ($ x10^{-6})	In Service PW ($ x10^{-6})	TOTAL IMPACT ($ x10^{-6})	Replacement Power Costs ($ x10^{-6})	Loss of Investment Costs ($ x10^{-6})	Site Cleanup Costs ($ x 10^{-6})	Total Avertable Costs ($ x10^{-6})	NET IMPACT ($ x10^{-6})
j	Δp_m	I_1	13.8 I_2	I_3	13.8 I_4	TI	I'_{51}	I'_{52}	I'_{53}	I'_5	NI

$TI = I_1 + 13.8\ I_2 + I_3 + 13.8\ I_4$

$I_5' = I_{51}' + I_{52}' + I_{53}'$

$NI = TI - I_5'$

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3.3 Methodology for the Value Analysis

3.3.1 The Content of the Value Analysis

The principal part of the Value Analysis is the estimation of the population dose, out to a distance of 50 miles, and the changes in dose due to the proposed Alternatives. The calculation is complicated by the fact that, for any one Alternative, the probability of a release of radioactive material in a given release category may change by a different amount from that in another category. Thus the population dose has to be estimated for each of several categories.

Other contributions to the value term are:

a) The averted radiation exposures to plant personnel that could occur as a result of severe reactor accidents, notably the dose that would be received by those engaged in the site cleanup operations.

b) The dose that would be received by plant and by contractor personnel in the installation phase of an Alternative. To meet NRC requirements this is treated as a negative value.

c) The changes in occupational dose during subsequent operation, following the incorporation of an Alternative. This could be either an increase or a decrease and thus could be a negative or a positive value.

3.3.2 Estimate of Avertable Population Dose

The method for evaluating this term is as follows: for each plant the conditional population doses, out to a radius of 50 miles, have been estimated (from CRAC-2) for each release category. For each of the release categories, doses corresponding to upper bound, central value, and lower bound values for the fractional releases of individual isotopes have been calculated. The upper bound value is based on the conservatively estimated fractions in the RSS (Reference 12). The central and lower bound values are 1/3 and 1/10 of the upper bound respectively. Thus in the unmodified plant the expected population dose would be

$$\sum^{\text{All } i} p_i \, D_i$$

where p_i is the probability of a release in Category i, giving rise to a 50 mile dose, D_i.

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0751

Given a modification which alters the various release probabilities for individual categories, we can write the new annual expectation as

$$\overset{\text{All } i}{\sum} p_i' D_i$$

where P_i' is the new probability of release in category i. We can write:

$$\overset{\text{All } i}{\sum} p_i D_i \quad - \quad \overset{\text{All } i}{\sum} p_i' D_i \quad = \quad \overset{\text{All } i}{\sum} \delta p_i D_i$$

where $\delta p_i = p_i - p_i'$.

Thus the change in the annual expectation due to a modification, or set of modifications, can be calculated directly, given the values of δp_i and D_i.

The total avertable dose is the sum of the changes in the annual expectations over the remaining life of the plant. Thus the factor to convert annual expectation to total avertable dose over plant life is 24 for St. Lucie Unit 1.

The present worth in dollars of the avertable dose is obtained by multiplying the annual averted dose by $1000 per person-rem and then discounting at 5% per year over the remaining life of the plant. The present worth factor is 13.8 for St. Lucie Unit 1.

3.3.3 Estimate of the Avertable On Site Dose

In the event of a severe reactor accident three possible causes of on site radiation exposure can be visualized:

a) Dose to operating personnel on site at the time.

b) Dose to personnel engaged in cleanup of the NPP and site.

c) On a twin-reactor site; increase in occupational dose to personnel working on the second reactor.

Each of these possible causes is discussed below.

a) Operating personnel on site at the time of the accident. It is considered that, with the possible exception of a few key personnel, an orderly evacuation of the site would take place prior to any release of activity. This cause would therefore not add significantly to the total exposure.

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b) Dose received during cleanup operations. In the absence of better data the "rule of thumb" of 40,000 man rem suggested by NRC (Reference 2) has been adopted.

c) Additional exposure of personnel on a multi-reactor site. An upper bound to this contribution can be estimated, for a twin reactor site, on the basis that operation of the other unit would be possible, from the radiological safety point of view, when the annual dose per person would not exceed 5 rem, instead of less than an average of 1 rem per person. Assuming that a total of 200 persons would be involved, the additional dose at the St. Lucie site would be about 800 man rem per year.

Assuming that such a dose could be received following an accident (anytime in the remaining plant life) which led to core melt, but not necessarily to a large release to atmosphere, it is discounted. Thus the maximum avertable dose would not exceed 11,000 p_m man rem, i.e., substantially less than the exposure assumed for cleanup operations.

To take account of the contributions at (b) and (c) the total avertable on site dose is assumed to be 51,000 p_m person-rem. The total averted onsite dose is the sum of the annual averted onsite doses over the remaining plant life, or 51,000 Δp_m x 24. In order to estimate the present worth of this term in dollars a monetary value of $1000 per person-rem is attributed to the averted exposure and then discounted at 5% over remaining plant life of 51000 x $1000 x Δp_m x 13.8.

Thus the maximum present worth of the avertable on site dose at the St. Lucie Unit lis estimated to be $\$7.04 \times 10^8\ p_m$ and the avertable cost for a change of Δp in p_m is $\$7.04 \times 10^8\ \Delta p_m$.

3.3.4 Estimate of Negative Values Associated with Changes in Occupational Doses.

These are estimated by the AE (see Section 3.2.2 above) but as indicated in Section 3.2.2 above, they are treated as Values.

3.3.5 Summary of "Value" Terms

The values for each proposed Alternative are summarized in tabular form using the format illustrated in Table L.3.

The format shows, for each Alternative, the Net Values for 3 cases corresponding to the upper bound, central and lower bound release fractions defined in Section 3.3.2 above.

The format shows also the fractional reduction in population dose and in total dose (i.e., population dose pluse on site doses) which would be achieved in each of the 3 cases for each Alternative.

0753

TABLE L.3-a ST. LUCIE UNIT 1 - SUMMARY OF VALUES (Based on Population Dose to 50 Miles, 5% Discount Rate)

		POSITIVE VALUES							
		Onsite		Offsite			Total		
Alter-native No.	Change in Core Melt Probability [Central Value]	Averted Dose (p-rem)	Present Worth Of Averted Dose @ $1000/p-rem ($ x10^{-6})rem	Averted Dose (p-rem)	Averted Dose + Base Case Dose	Present Worth of Averted Dose @ $1000/p-rem ($ x10^{-6})	Averted Dose (p-rem)	Averted Dose + Base Case Dose	Present Worth of Averted Dose @ $1000/p-rem ($ x10^{-6})
					[U, C, & L Values]			[U, C, & L Values]	
j	Δp_m	V_1	V_1'	V_2	ADR_o	V_2'	V_{12}	ADR_n	V'_{12}

V_1 = 51000 x Δp_m x 24
V_1' = 51000 x Δp_m x $1000 x 13.8
V_2 = Averted Dose x 24
V_2' = Averted Dose x $1000 x 13.8
ADR_o = V_2 + (Basecase Dose x 24)
V_{12} = $V_1 + V_2$
ADR_n = V_{12} + (Basecase Dose x 24 + V_1)
V_{12}' = $V_1 + V_2'$

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TABLE L.3-b ST. LUCIE UNIT 1 - SUMMARY OF VALUES (Based on Population Dose to 50 Miles, 5% Discount Rate)

		NEGATIVE VALUES							
		Installation		Operation		Total		Net Value	
Alternative No.	Change in Core Melt Probability [Central Value]	Installation Dose (p-rem)	Present Worth of Installation Dose @ $1000/p-rem	In-Service Operational Dose (p-rem)	Present Worth of In-Service Oper. Dose @ ($ x10^{-6})	Installation and Operational Dose	Present Worth of Install. & Oper. Dose @ $1000/p-rem ($ x10^{-6})	Averted Dose (p-rem)	Present Worth of Averted Dose @ $1000/p-rem ($ x10^{-6})
								[U, C, & L Values]	
j	Δp_m	V_3	V_3'	V_4	V_4'	$V_3 + V_4$	$V_3' + V_4'$	NV	NV'

V_3 = V_3 x $1000
V_4' = $(V_4 + 24)$ x $1000 x 13.8
NV = $V_1 + V_2 - V_3 - V_4$
NV' = $V_1' + V_2' - V_3' - V_4'$

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3.3.6 Special Situations Affecting the Value Analyses

As indicated in Section 2.1 above, the NRC has emphasized the need to examine the conditions at each NPP to determine whether there are any special situations that would increase the off-site effects of a severe reactor accident.

In the case of the St. Lucie NPP, its location on the shore of the Atlantic Ocean has two effects:

a) It introduces the possibility that deposition of activity from the plume into the ocean could create additional hazards due to radioactive contamination of fish.

b) If the population dose is based on an integration out to a limit of 50 miles, this tends to reduce the weather averaged dose by a factor of about 2 since in about one half of the possible wind directions most of the plume's travel would be over water.

A recent international collaborative study of the effects of activity deposition from reactor accidents into liquid pathways (Reference 13) has shown that the effect of these would be small compared with the effects of deposition over land.

3.4 Methodology for the Value-Impact Analysis

Once the results of the Impact analysis and the Value analysis have been summarized on Tables L.2 and L.3, as described above, the Value-Impact analysis itself is almost self evident. For simplicity of interpretation, the results are presented in the format shown in Table L.4. This provides for the following:

a) The analysis based on off site costs only is shown separately from the analysis based on off site and on site costs only.

b) The results presented in Table L.4 are based on "Best Estimate" values for the various impact parameters coupled with upper, central and lower bound estimates on source terms.

c) The predicted reduction in probability of core melt (Δp_m) and the predicted reduction in total dose (population and occupational) are shown for each Alternative. The fractional reduction in total dose is also shown.

d) The Value-Impact relationships are shown in 3 different ways for each Alternative, namely:

i) Dimensionless Value/Impact ratio,

ii) Net benefit,

TABLE L.4 ST. LUCIE UNIT 1 - SUMMARY OF VALUE-IMPACT ANALYSIS (Based on Population Dose to 50 Miles; 5% Discount Rate)

		V-I ANALYSIS BASED ON OFFSITE COSTS							V-I ANALYSIS BASED ON OFFSITE AND ONSITE COSTS				
						Measures of V-I					Measures of V-I		
Alternative No.	Change in Core Melt Probability [Central Value]	Offsite Averted Dose (p-rem)	Total Averted Dose + Base Case Dose	Total Impact [Central Cost] ($ $\times 10^{-6}$)	Present Worth of Averted Dose @ $1000/p-rem [U,C,&L Values] ($ $\times 10^{-6}$)	V-I Ratio	Net Benefit ($ $\times 10^{-6}$)	Dollars per p-rem	Net Impact [Central Cost] ($ $\times 10^{-6}$)	Present Worth of Averted Dose @ $1000/p-rem [U,C&L Values] ($ $\times 10^{-6}$)	V-I Ratio	Net Benefit ($ $\times 10^{-6}$)	Dollars per p-rem
		[U, C, & L Values]											
j	Δp_m	V_2	ADR_n	TI	V_2'	VIR_o	NBV_o	DPR_o	NI	NV'	VIR_n	NBV_n	DPR_n

$VIR_o = V_2' \div TI$
$NBV_o = V_2' - TI$
$DPR_o = TI \div V_2$

$VIR_n = NV' \div NI$
$NBV_n = NV' - NI$
$DPR_n = NI \div NV$

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iii) Dollars expended per person-rem of population dose averted.

The central estimate value-impact results are also summarized in Table L.5.

TABLE L.5 ST. LUCIE UNIT 1 - SUMMARY OF VALUE-IMPACT MEASURES (Central Value)

				V-I Analysis Based on Offsite Costs			V-I Analysis Based on Offsite and Onsite Costs		
Alternative No.	Change in Core Melt Probability	Offsite Averted Dose p-rem	Averted Dose Ratio	V-I Ratio	Net Benefit ($ x10^{-6})	Dollars per p-rem	V-I Ratio	Net Benefit ($ x10^{-6})	Dollars per p-rem
j	Δp_n	V_2	ADR_n	VIR_o	NBV_o	DPR_o	VIR_n	NBV_n	DPR_n

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0759

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This is one of six case studies for USI A-45 Decay Heat Removal (DHR) Requirements. The purpose of this study is to identify any potential vulnerabilities in the DHR systems of a typical Combustion Engineering PWR to suggest possible modifications to improve the DHR capability, and to assess the value and impact of the most promising alternatives to the existing DHR systems. The systems analysis considered small LOCAs and transient internal initiating events, and seismic, fire, extreme wind, internal and external flood, and lightning external events. A full-scale systems analysis was performed with detailed fault trees and event trees including support system dependencies. The system analysis results were extrapolated into release categories using applicable past PRA phenomenological results and improved containment failure mode probabilities. Public consequences were estimated using site specific CRAC2 calculations. The Value-Impact (VI) analysis of possible alternatives considered both onsite and offsite impacts arriving at several risk measures such as averted population dose out to a 50-mile radius and dollars per person rem averted. Uncertainties in the VI analysis are discussed and the issues of feed and bleed and secondary blowdown are analyzed.

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